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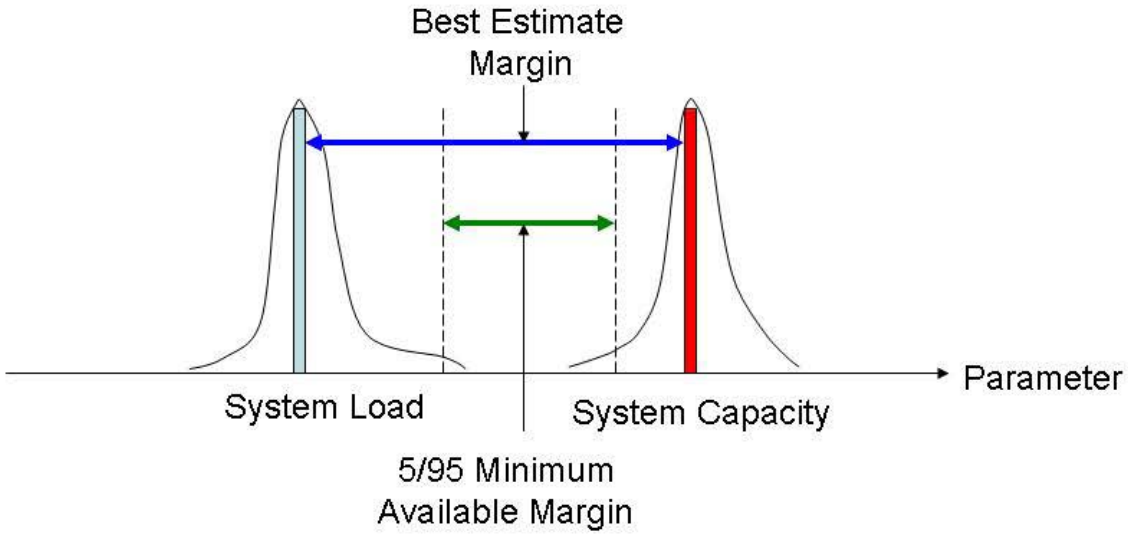
# Framework for Risk-Informed Safety Margin Characterization



# Framework for Risk-Informed Safety Margin Characterization

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# PRODUCT DESCRIPTION

Commercial nuclear power plants in operation continue to undergo design and operational changes to support cost-effective long-term operation. Additionally, as plant operational lifetimes are extended, it is imperative that they effectively manage aging degradation, prevent the occurrence of any safety-significant operational events, and analytically demonstrate acceptable (and even improved) nuclear safety risk. This report describes initial research to develop and validate an integrated framework and advanced tools for risk-informed safety analyses that will enable accurate characterization and visualization of nuclear power plant safety margins.

## Results and Findings

This research evaluated the current state-of-the-art with respect to research into nuclear power plant safety margins. It also reviewed tools in widespread use to perform plant safety analyses and probabilistic risk assessments for their applicability to address safety margin issues. Currently, no consensus approach exists to evaluate and manage nuclear power plant safety margins that is both accurate and efficient. For performance of safety margin evaluations, the current generation of safety analysis and probabilistic risk assessment tools is not sufficiently integrated to permit efficient analysis. In addition, enhancements to some safety analysis tools will be necessary to permit evaluation of postulated plant enhancements that could be implemented to support long-term plant operation.

## Challenges and Objectives

Because of a number of important societal issues, it is imperative that the current fleet of commercial nuclear power plants continues to operate at high performance levels. Although the current level of performance of the fleet is excellent, it is not certain that it is sustainable over the future periods of extended plant operation that are envisioned. The original licensing of commercial nuclear power plants established adequate safety margins by performing conservative engineering analyses and using conservative judgment in specifying appropriate safety limits for critical plant parameters. Over time, however, plant operation has the potential to affect the original design margins. Operational changes made to enhance plant economics, such as power uprates, can also have an impact on safety margins. Although technically sound, current analytical approaches and tools used to assess safety margins are inefficient and labor-intensive to apply. In addition, technological advances being pursued to enhance long-term plant operation may require analysis outside the bounds under which some of the current safety analysis models are valid. This research task is intended to address these issues by developing a consensus approach to performing safety margin evaluations, with eventual development of an integrated suite of analytical methods and tools that support their effective and efficient implementation.

## Applications, Value, and Use

Maintenance of safety margins has served as a foundational principle of plant operation and regulation since the advent of commercial nuclear power. However, to date, minimal research has been conducted on methods and tools to monitor safety margins over the entire plant life cycle. The most comprehensive research in this area has been performed by the Nuclear Energy Agency Committee on the Safety of Nuclear Installations Safety Margins Working Group. A review of this research indicated that application of existing methods and tools to address practical safety margin issues would be time-consuming and inefficient. The results presented in this report provide a useful foundation on which an integrated analytical approach and suite of application tools can be developed to permit plants to evaluate and manage safety margins over the full lifetime of plant operation, including operation during license extension intervals.

## EPRI Perspective

The evaluation and maintenance of plant safety margins will be an important element in enabling the long-term operation of the current fleet of commercial nuclear power plants. An important element of this evaluation will be to develop a consensus method and integrated suite of tools to facilitate the effective and efficient performance of safety margin evaluations. This research report provides a critical initial step to identify and prioritize enhancements of methods and tools to achieve this objective.

## Approach

This research evaluated existing tools in widespread use to perform nuclear plant safety analyses and probabilistic risk assessments. These tools were characterized from the viewpoint of their capabilities to support safety margin assessments.


## Keywords

Probabilistic risk assessment
Safety analysis
Safety margins

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# 1
# INTRODUCTION

Due to a number of important societal issues, it is imperative that the current fleet of commercial nuclear power plants (NPPs) continues to operate at high performance levels. These societal benefits include the following:

1. Nuclear power is free of greenhouse gas (GHG) emissions. To meet proposed GHG reduction targets, EPRI studies show that the existing fleet of NPPs must continue to operate at the current levels of high performance at least through the year 2030.
2. The currently operating fleet of NPPs provides inexpensive electricity to the consumer. Most of the plant's capital cost is already paid. Additionally, operating costs are low due to the relatively low cost of nuclear fuel. These costs would continue to be relatively low even with much higher uranium prices and with large capital improvements implemented in the plants.
3. Current NPPs provide future energy security in light of projected demand growth of electricity consumption.

In addition to the societal benefits, there also are clear financial benefits to owners and operators of the current fleet of NPPs as well. Thus, the Long Term Operability (LTO) challenge is to assure the continued operation of the current fleet of NPPs worldwide at the current high performance levels through 2030 and beyond.

Although the current level of performance of plants is excellent, it is not certain that it is sustainable over the long operational periods that are envisioned. Modern industrial society does not have many examples of operating complex facilities at high levels of performance for 60, 80, or more years. Generally, installations that remain serviceable over such timeframes are overtaken by facilities with new technology with better performance and lower costs. Thus, LTO for the existing fleet of commercial NPPs represents a unique challenge.

Surveys, workshops, and literature reviews have identified a number of technical issues and opportunities to address the LTO challenge. Some of these areas address specific risks to long term operation. In some cases research will be necessary to develop approaches to mitigate the risks. In other cases the desired research will result in analyses or inspections that demonstrate that the risk is below a level where it is of significant concern. Finally, some research will be directed at providing methods to confidently manage the risk and to schedule and implement capital improvements. Others technical areas will present opportunities to enhance and modernize the plant to support achieving sustained high levels of performance. In these cases, the research would include consideration of the cost to benefit of such improvements.

Risk-informed regulations and operations of nuclear power plants have become part of regulatory policy of the U.S. Nuclear Regulatory Commission (NRC) and of the operational culture at NPPs in the United States. That this would be the case is both logical and technically sound. First, elements of nuclear plant licensing, design, and operation originally were established when there was little operating experience with nuclear plants. As such, licensing requirements were conservative and prescriptive with designs that possess significant safety margin. With more than 3000 operating years of experience in the U.S. alone, NPP operators confidently can optimize these requirements and processes. Two decades of carefully implementing risk-informed processes have demonstrated that nuclear safety risk is an appropriate criterion for such optimization of regulations and operations. This optimization simultaneously can improve safety, plant performance, and cost to both the plant operators and to electricity consumers. With considerations for differences in safety policies and practices, risk-informed operations and regulations also can be adapted for application world-wide.

## 1.1 The Concept of Safety Margin

The concept of safety margin is one that is familiar to the design of structures in civil engineering. In the standard approach, a system is designed to possess a "capacity" which is specified such that it is sufficient to withstand any postulated "load" which the system is envisioned to experience during service. In this approach, the system is designed so that there is sufficient "margin" so that the likelihood of the load exceeding the system capacity to handle it (due to uncertainties in the capacity, load or both) is very small. This concept is shown schematically in Figure 1-1. In this framework, the safety margin is the difference between the characteristic value of the "capacity" and the characteristic value of the "load." Note that both the capacity and load are represented by distributions and thus there exist uncertainty in their "true" values. In this approach, the load and capacity are each characterized by a statistical distribution. Thus margin can be defined in several ways. One such definition is the difference between the best estimates of each quantity (shown as "Best Estimate Margin" in Figure 1-1). However, because both the load and capacity are characterized by statistical distributions, there is some possibility that the load could exceed the system capacity in certain situations. Thus, another measure of margin is to determine the difference between two convenient points in the distribution. An example of this is shown in Figure 1-1 as the "5/95 Minimum Available Margin" because it uses a (nearly) worst case scenario using the 5% left tail of the capacity distribution function and the 95% right tail of the load distribution function. We note that this definition of margin is conservative and its calculation requires a significant amount of data (for both the load and capacity) to permit characterization of the distribution functions.

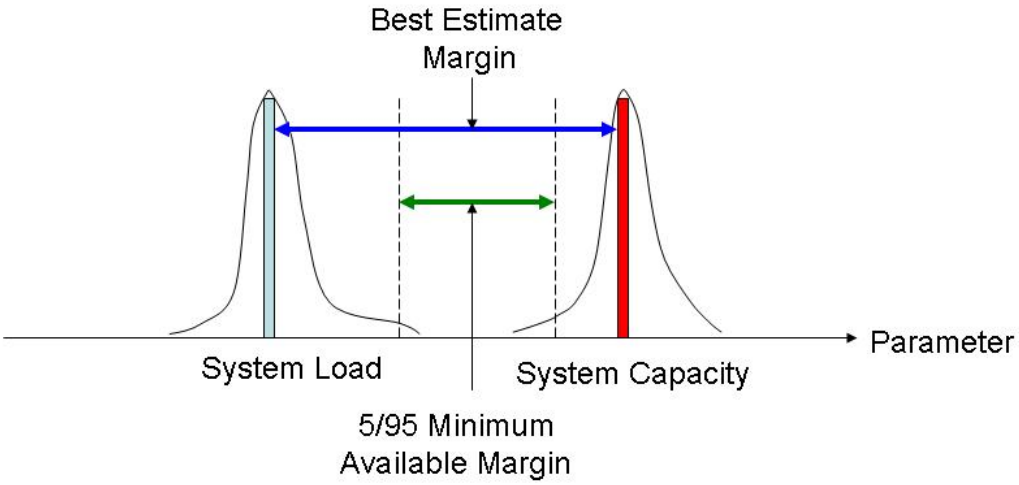


**Figure 1-1**
**Safety Margin Concept**

However, during the design and licensing of the current fleet of commercial NPPs, actual data from which estimates of the actual capacities and loads for plant systems, structures and components (SSCs) could be developed were either not available or too expensive to obtain. Thus, alternative methods were devised to ensure sufficient safety margins were built into the plant design and operational framework. The first approach was to specify plant designs that were very conservative from an engineering perspective. This design conservatism resulted in utilizing SSCs that are capable of performing at levels that are significantly higher than what is required to support normal operation, envisioned plant transients or design basis accidents. The second approach addressed the issue from an operational perspective by specifying a "hard" safety limit which is set at a level that is significantly below the designed system capacity (see Figure 1-2).

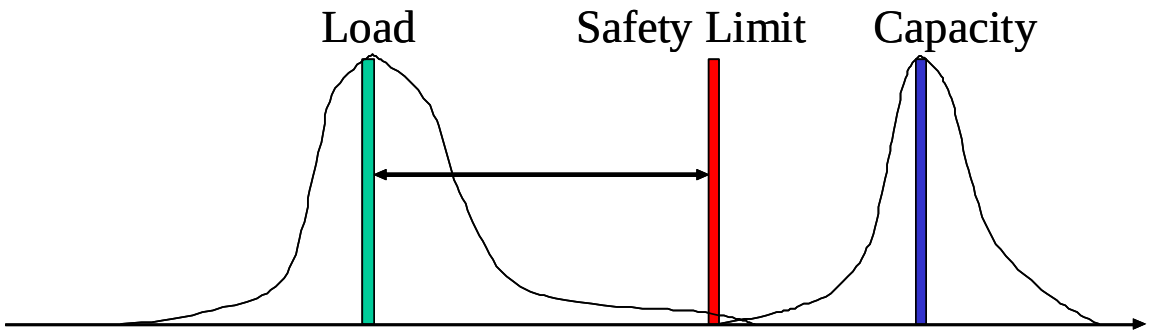


**Figure 1-2**
**Safety Limit Concept Applied to Nuclear Power Plant Licensing**

## 1.2 Impacts of Long Term Operation on Safety Margins

In the licensing of commercial NPPs, the specification of an adequate safety margin was accomplished by a combination of performing conservative engineering analyses to estimate the system loads and using similar conservative judgment in specifying appropriate safety limits for critical plant parameters. From a licensing perspective, these margins are specified by specification of limits on plant parameters important to ensure nuclear safety; in particular parameters that, if not exceeded, provide assurance that the principle barriers to fission product release (that is, the fuel, the reactor vessel and primary system piping, and the containment) successfully will perform their intended function. In the United States regulatory system, these limits are specified either in federal law or in the plants Technical Specifications. A listing of such limits is provided in Appendix A for reference.

Over time, however, NPP operation has the potential to impact the original design margins. This can occur by changing the expected value or the distribution function for the load or the capacity. For example, ageing of plant materials can result in decreased resiliency of the system to withstand perturbations; thus causing the capacity curve to shift to the left (as shown in Figure 1-3). Additionally, operational changes made to enhance plant economics also can impact safety margins. For example, increased fuel burnups and plant power uprates can result in operation of plant SSCs at higher stress levels (that is, closer to their design tolerances) and shift the load curve to the right. However, note that not all actions taken will result in decreases in safety margins. For example, some PWR plants have installed dedicated reactor coolant pump seal injection systems. For these plants the likelihood of an RCP seal LOCA is greatly reduced and thus margins enhanced. As another example, improved analytical methods and supporting operational data can provide improved estimates of actual SSC performance; thus shifting the expected value of the load curve to the left. As a final example, the implementation of diagnostic condition based maintenance activities (such as vibration monitoring or lubricating oil analysis) provide an effective means of identifying degraded conditions of rotating equipment at an incipient stage. As a result, these technologies influence the actual safety margins by decreasing the variance of the distribution of the load function; thus providing a higher degree of confidence that safety margins are being maintained.

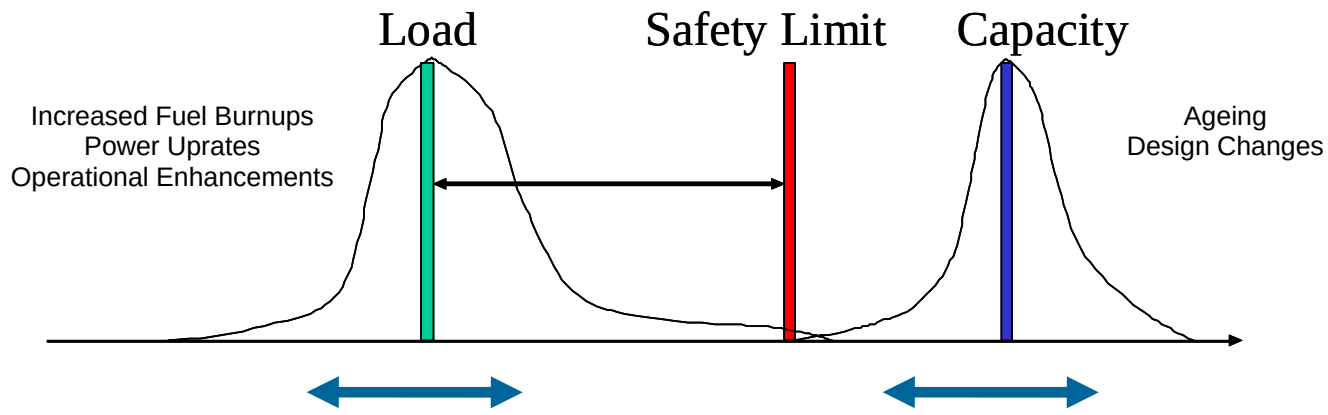


**Figure 1-3**
**Impact of Plant Long Term Operation on Safety Margin**

## 1.3 Summary of Previous Safety Margin Research

Maintenance of safety margins has served as a foundational principle of NPP operation since the advent of commercial nuclear power. As was discussed in Chapter 1, the need to ensure adequate margins were maintained led to specification of limiting conditions for operation (LCOs) for critical plant parameters in the plants Technical Specifications. In addition, regulators have consistently included consideration of the potential impact on safety margins as part of their reviews.

In 2003 the Nuclear Energy Agency Committee on the Safety of Nuclear Installations (NEA/CSNI) formed a working group to evaluate the impact of commercial nuclear power plant (NPP) life extension, aging and operational changes on plant safety margins. This task group consisted of senior scientists and engineers responsible for safety technology representing regulatory authorities from several nations. Recently, this group published their final report on development of a Safety Margins Action Plan (SMAP) [1] that addresses five activities:

- Develop a working definition of safety margins and related concepts
- Develop a process for the assessment of safety margins
- Identify appropriate methods for safety margin evaluation
- Identify methods for safety margin quantification
- Prepare a CSNI guidance document on safety margins for use by NPP regulatory authorities

In this section, we provide a summary of the results and conclusions contained in the SMAP final report.

The SMAP Task Group was formed as a result of conclusions reached by the NEA/CSNI Committee on Nuclear Regulatory Activities which identified the potential for actions taken by NPP operators to achieve economic or operational enhancements (for example, power uprates, increased fuel burnups, and so on) to erode plant safety margins as specified in the original

licensing basis of the plant. The situation is complicated by the possibility that, although an individual design or operational change may not result in a significant erosion of safety margins, the cumulative effect of multiple changes may result in a challenge to them. Additionally, the SMAP Task Group also recognized the significant advances made in safety analysis techniques, both deterministic and probabilistic, should be incorporated into the analysis and specification of safety margins. The key issue identified by the SMAP Task Group was that the two approaches to performing NPP safety analysis (that is, deterministic and probabilistic safety analysis) utilize different approaches that were developed relatively independently of each other. As a result, obtaining consistent analysis results when both methods are employed has been found to be a challenge. As a result of the above objectives, the SMAP Task Group developed a proposed methodology to evaluate NPP safety margins. The proposed approach is based on a combination of classical deterministic safety analysis (DSA) and probabilistic risk assessment (PRA) methods. In the SMAP Final Report [1], the proposed method was applied to two examples to demonstrate the applicability of the concept.

Historically, the approach to managing safety margins applied to commercial NPPs consisted of defining an applicable safety limit and ensuring plant operating conditions, including during postulated design basis accidents, remain below the specified limits. To ensure the safety of NPP operations over both normal and postulated accident conditions, the safety limits were specified by including varying degrees of conservatism (which usually were large) in the supporting analyses. However, this approach has been subject to several criticisms. First, because the physical models on which the safety limits were based represented (in some cases crude) approximations, they could not provide verifiable estimates of actual plant behavior over all possible operational situations. Second, because multiple analytical methods and computer codes were used in the specification of the safety limits, individual conservatisms were combined in their specification. Due to nonlinearities in the underlying physical processes, it could not be proven that the combination of the individual conservatisms would result in a conservative prediction of plant response. Finally, the initial specification of NPP safety margins relied on the use of bounding representative accident classes. Because the Three Mile Island Unit 2 accident demonstrated that accidents considered less serious than the bounding design basis accident could result in significant core damage, the concept of safety margin began to be more broadly interpreted and applied. This broader application of safety margins was one of the major motivations for formation of the SMAP Task Group.

The traditional setting of plant safety margins has been applied at both the barrier and system levels. Thus, in the interpretation of the SMAP Task Group, a NPP will possess many safety margins that need to be specified and managed. In this framework, the safety margin is developed by adding individual margins to achieve a “global plant margin.” The SMAP Task Group identified the following types of margins which contribute to the global plant margin (see Figure 2-1 on p. 22 of reference [1] for a graphical depiction):

- Analytical Margin
- Licensing Margin
- Barrier Margin

- Source Term Analytical Margin
- Source Term Margin
- Dose Margin

The first two margins are viewed as addressing individual safety variables to specify appropriate acceptance criteria. The analytical margin is viewed as the margin provided to address actual ("real") transients anticipated to be encountered whereas the licensing margin is viewed as the additional margin provided to ensure adequate protection against the enveloping transient. The next three items address margin in the source term for radioactive material release. The first of these (that is, the third bullet) addresses the margin in the barrier integrity limit. Similar to the first two margins listed, the source term margins also are classified into analytical (that is, margin provided to address "real" transients) and source term (that is, additional margin in the enveloping transient). Finally, there is additional margin in the estimated radiological dose received by the public.

As can be seen, this proposed approach represents a comprehensive, but complex, approach to characterizing safety margins. This is because the different margins address fundamentally different physical effects; thus there will be a significant challenge in any activity which attempts to achieve an integrated characterization.

To permit a comprehensive assessment of NPP safety margins, the SMAP Task Group developed a proposed integrated structure for the identification and analysis of applicable safety margins. The approach is based on the multiple levels of barriers to release of radioactive materials present in commercial NPPs. To risk-inform the proposed approach, the acceptable limits for safety margins respective to each of the barriers is conditional to the likelihood of the underlying event(s) against which the barrier provides protection. For purposes of illustration, the SMAP Task Group utilized the categorization proposed by the International Atomic Energy Agency (IAEA) in reference [2].

- Category 1: transients related to normal operation
- Category 2: incidents of moderate frequency ($> 10^{-2}$/ry)
- Category 3: very low frequency accidents ($10^{-2}$/ry to $10^{-4}$/ry)
- Category 4: hypothetical accidents ($10^{-4}$/ry to $10^{-6}$/ry)

**Note**: ry = reactor years of operation. As the likelihood of the initiating event decreases, the level of consequences that is acceptable increases. As a result, the concept is very similar to use of a frequency – consequence curve to specify acceptable performance limits.

As an example, we provide a brief summary of how this classification would be applied to ensuring the protection of the nuclear fuel, which serves as the first barrier to fission product release. The major threats to this barrier are thermal and mechanical loading on the cladding and the potential for melting of the fuel pellets due to overheating. Thus, specific criteria are

specified which are intended to prevent those damage mechanisms (in the SMAP Task Group report, these are called decoupling phenomena). For protection of fuel, these result in limits on the following:

- Prevention of large cladding temperature increases (by specification of critical heat flux limits)
- Prevention of fuel melting (by specification of a limit on fuel maximum linear average power)
- Prevention of cladding embrittlement (by limiting maximum fuel cladding temperature)
- Prevention of mechanical failure (by limiting relevant mechanical properties of the fuel such as cladding circumferential deformation, fuel rod internal pressure, and so on.)

The frequency of events that can occur in the two lowest categories of the IAEA classification scheme is relatively high. Thus, the applicable safety limits are set stringently to ensure there are no unacceptable radiological consequences associated with any events in either of these categories, should they occur. This is achieved by setting fuel parameter limits such that the occurrence of a Category 2 event will have no effect on the fuel cladding barrier.

Because the likelihood of the occurrence of a Category 3 event is much less than for Category 1 or 2, limited damage in some fuel rods would be permitted. However, the criteria would be set so that any damage would not degrade the ability to provide long-term core cooling. Finally, Category 4 events are characterized as having a low probability of occurrence, examples of which include large break loss of coolant accidents (LB LOCAs), main steam line break (MSLB) accidents and reactivity insertion accidents (RIAs). In this category, significant damage of a few fuel rods would be allowed as long as the long-term core cooling capability is maintained. However, because events in this category can produce severe stresses on the fuel barrier, specific postulated event sequences need to be analyzed to determine the need for additional safety margins (that is, in addition to the generic parameters applicable to the other three categories) to address event specific issues. Examples of these types of margins could include the specification of maximum cladding oxidation and peak cladding temperature that would be expected to occur during a LB LOCA.

In the approach proposed by the SMAP Task Group, similar analyses are performed and safety margins specified for the other two barriers, the primary coolant system (PCS - called the primary circuit in the SMAP report) and the containment system. The SMAP task group identified the primary risks for failure of the PCS barrier as thermal and mechanical loading. This is achieved by limiting PCS pressure and thermal cycles. For the containment, the SMAP Task Group identified both thermal and mechanical loading as primary risk factors; however, they also identified radiation induced leakage as an additional risk factor for which applicable safety margins must be specified.

As a result of this analysis, the SMAP Task Group developed a proposed methodology for the assessment of safety margins for the purposes of NPP regulatory decision-making. A fundamental principle for the proposed process is that a satisfactory assessment of NPP safety margins for a particular application requires consideration of all possible scenarios that have a non-negligible likelihood of occurrence. The SMAP Task Group defined this set of scenarios as

the "risk space" that is described in an event tree format similar to that used in a plant PRA model. The development of the risk space requires specification of a complete set of event initiators and corresponding event trees for their analysis. A "base case risk space" represents the current operational state of the NPP and is used to compare any proposed changes. We note that this approach is consistent with that employed in the United States for risk-informed regulatory applications (as described in [3]). However, in the opinion of the SMAP Task Group, the current PRA structure possesses several limitations which require enhancement. First, the use of binary end states representative of current PRA models (for example, either the end state results in core damage or it does not) does not address the issue of how much margin exists between the physical end state condition and the state of unacceptable damage. Second, current application of PRA technology only analyzes end state consequences that result in severe NPP damage states (that is, core damage and/or large early release). Thus the application of PRA currently does not analyze scenarios that can result in less severe (but potentially more frequent) releases of radioactive materials to the environment.

As mentioned previously, the primary objective of the SMAP Task Group was to investigate methods to perform a regulatory analysis of the potential impact of NPP modifications or operational changes on safety margins. In the SMAP final report the task group identified several additional limitations associated with existing PRA technology to achieve this objective. For this purpose, the primary figure of merit for decision-making is the frequency that a safety margin could be exceeded. With respect to this metric, the SMAP Task Group concluded that the timing of actions will be a significant factor; thus consideration of timing of events will need to be an essential element of the analysis process. The group also determined that for a PRA to capture conditions that could result in a change in safety margins, the evolution of the plant conditions would specify the path the event follows. The SMAP Task Group called these event branch points' stimuli. In the viewpoint of the group, the stimuli condition the events. Thus, the history of the event also is an important determinant in calculating its frequency of occurrence. The SMAP Task Group concluded with a recommendation to investigate the use of the "stimulus driven theory of probabilistic dynamics" [4] as a potential method to address the issue.

The conclusion of the SMAP Task Group was to extend PRA methods to the full risk space so that it would possess the capability to address (at least) the full set of licensing objectives considered in the NPP licensing basis. To this end, they proposed that the set of conditions specified in 10CFR50.59 serve as a starting point for evaluating the applicability of proposed changes that impact NPP safety margins. To achieve this objective, the SMAP Task Group proposed the following modifications to make the process applicable to the evaluation of safety margins:

- Expansion of the selected set of initiating events (IEs) considered in 10CFR50.59 evaluations (that is, analyze a comprehensive set of IEs beyond design basis accident and abnormal operational occurrences).
- Expansion of the analysis scope to include SSCs and operator actions credited in the PRA and to expand the scope of consequences to address potential failure modes. In both cases, in the method proposed by the SMAP Task Group the extent of what would need to be included

in the analysis would be dependent upon the particular application. Note that in the proof of concept cases studies; this represented a significant expansion of the risk space from that addressed in a typical PRA.

- Application of 10CFR50.59 questions concerning changes in consequences should be interpreted in terms of changes in exceedance probabilities.

Some brief elaboration should be made on the expansion of the analysis and consequence scope in the second bullet given above. As mentioned previously, the proposed method was performed on two proof of concept applications by participants on the SMAP Task Group. In the first example, the question of a plant modification to increase the emergency core cooling system (ECCS)/containment spray sump strainer size was evaluated. The safety margin that was evaluated was the impact on the net positive suction head (NPSH) for the respective pumps. In this application, the questions posed to perform a 10CFR50.59 safety evaluation were modified to determine how PRA event trees would need to be modified as a result of the modification. In this demonstration, a phenomena identification and ranking table (PIRT) was used to calculate the conditional probabilities of loss of function for each event sequence identified in the risk space. A key observation from this application was that the process of generating the risk space was iterative with the identification of the key safety variables. The figure of merit for regulatory decision-making was the calculation of the total change in core damage frequency ($\Delta$CDF) from the conditional probability of loss of NPSH and change in event sequence frequency. The second example consisted of the evaluation of changes in peak fuel cladding temperature (PCT) due to a power uprate and to evaluate its impact on plant safety risk. In this case, the evaluation of the physical parameters was significantly more complex that for the NPSH case study. In this case, a PIRT was not performed to evaluate the system performance; rather, a RELAP5 thermal hydraulics model was constructed for this purpose. Additionally, in this example multiple margins were evaluated (for example, PCT margin for small break LOCAs was evaluated separately from the margin for large break LOCA scenarios). A key conclusion that can be drawn from these case studies is that, although the proposed process can provide useful and informative results for the evaluation of the impact of plant changes on safety margins, it appears that it will be very complex and labor intensive to perform on actual applications.

## 1.4 EPRI Long Term Operation Safety Margin Research Plan

The vision for the Risk-Informed Safety Margin Characterization (RISMC) research of the EPRI LTO initiative is to develop an integrated approach and suite of supporting implementation tools to permit cost-effective safety margin assessments that address the challenges and opportunities associated with long term NPP operation. A major focus of this research is to address the limitations associated with technology currently employed in the safety analysis and risk assessment of operating NPPs. In particular, the current methods and application tools constitute a “brute force” approach that requires a sequence of series activities to evaluate and characterize safety margins. As a result, the current state of the art is labor intensive, time consuming and expensive to perform. A secondary objective of the research will be to address enhanced RISMC capabilities to address future challenges and opportunities beyond those currently addressed [5].

The planned research and development can be viewed as consisting of three interrelated activities addressing the following issues: (1) advanced PRA methods, (2) advanced DSA

methods, and (3) methods/tools for analysis integration and visualization of results to support effective and efficient risk-informed decision-making. For all three groups, the research is intended to address analysis completeness, treatment of uncertainty, and computational efficiency.

In the PRA area, development of advanced analysis techniques continues to be an area of ongoing research and development. From the perspective of NPP owner/operators, significant effort has been expended to obtain practical risk-informed operational applications. A significant example of this in the United States is the broadening applications of configuration risk management to plant operations and maintenance programs in support of both regulatory requirements (that is, the Maintenance Rule 10CFR50.65(a)(4)) and operational/economic considerations. There also continues to be significant ongoing research to improve the computational efficiency of risk assessment and to develop methods and tools that more accurately characterize and address the limitations inherent in the current state of the art. Examples include investigation of techniques such as declarative modeling, direct probability calculation and binary decision diagrams [6].

While important advances are being made to improve the state of the art in PRA technology, its effective use in the decision process has been limited. A significant limitation that is directly applicable to the management of safety margins is the reliance on current generation DSA tools to determine appropriate PRA success criteria. This is due to the fact that the current generation of DSA methods and tools does not support a framework to account for uncertainty in the deterministically estimated values of "load" and "capacity." For example, where margins are relatively small or where large uncertainties exist, it is possible for functional success to occur in the PRA event tree's failure branch, and conversely, it is possible for a functional failure to occur in the success branch. A particularly important example is the application of PRA to passive systems [7, 8]. Another example of a limitation of the current state of the art in PRA methods is the assumption of independence of failure rates on the system's state and evolution. Techniques for dynamic PRA, which are intended to address this issue, are still in a very early phase of development.

In the DSA area incremental advances to improve modeling of plant components and transient/accident phenomena have been made over the past two decades. However, the analysis tools currently in use are based on a modeling framework and computational methodology that was developed nearly 40 years ago. Thus, the advantages of modern developments in computer technology/computational science and engineering have not been utilized. Although the current suite of codes has served as an adequate basis to address safety margin analysis, significant enhancements will be necessary to support the challenges of extended plant operational life cycles. These challenges include issues such as the presence of errors in numerical approximations and methods to integrate the results obtained from the distinct physical models employed at the different stages of analysis. The use of individual physics-based models (that is, thermal hydraulics, neutron kinetics, and so on) and their explicit coupling to simulate reactor transient results in an inefficient approach to DSA. The current methods and tools also fail to capture complex multi-dimensional and tightly coupled multi-physics behavior.

From the perspective of a NPP owner/operator, the vision for the RISMC research is to develop an integrated approach and implementation tools that permit cost-effective safety margin assessments that are capable of effectively and efficiently addressing the challenges and

opportunities associated with extended NPP operation. This is intended to include development of enhanced RISMC capabilities to address future challenges and opportunities beyond those currently envisioned. There are several critical elements to achieving this objective. Thus, the objectives of the RISMC research plan are intended to address these issues over a three to five year time horizon as described below.

1. First, any approach developed for RISMC must provide technically accurate results that are accepted by all involved stakeholders. Thus, an important element of the RISMC research plan is to develop a consensus approach that can be accepted by nuclear plant operators, regulatory authorities, and other stakeholders for risk-informed safety margin assessments. Thus, an objective of the RISMC research plan is to make significant progress on the development of such a consensus approach.
2. The utility of an approach to evaluate and manage safety margins is in its application to address important issues associated with extended plant life cycles. It should be noted that these applications of the approach are not limited solely to addressing issues associated with ageing management. Application of the approach also will be necessary to evaluate opportunities for enhanced operation. Thus, an important research task will be to identify appropriate "case studies" that can be used to validate the RISMC methodology and tools that are developed. Several examples of risk-informed applications that could be used to achieve this objective include:
   - Evaluation of design basis requirements for large-break Loss of Coolant Accidents and/or High Energy Line Breaks.
   - Optimizing operations of pressurized water reactors during mid-loop conditions in the shutdown mode.
   - Evaluating the broad application of risk-managed Technical Specification Allowed Out-of-Service Times.
   - Performing a safety assessment of a phased, multi-cycle plant refurbishment and power up-rate.
   - Evaluating application of risk and safety monitoring that utilizes real-time operating parameter data, equipment configurations, success criteria, and reliability information.
   - Considering application of an efficient Significance Determination Process assessment to evaluate emergent safety issues.

   Thus, an objective of the RISMC research program will be to identify one or more representative applications and perform case studies to demonstrate the technical adequacy of the methods developed to perform safety margin characterizations.

3. A third critical element of the RISMC research will be to develop an integrated suite of advanced tools to efficiently conduct the technical assessments. Specifically, the RISMC research program has the following objectives:
   - Through collaboration with other stakeholders make significant progress on the development of enhanced safety analysis (DSA) capabilities, computational engines, results visualization, and validation.
   - Achieve broad application of an integrated probabilistic risk assessment (PRA) tool with advanced computational methods, full scope PRA aggregation capabilities, results visualization, and connectivity to plant information to support configuration risk management.
   - Progress toward an integrated RISMC capability including interface of PRA and DSA codes, broad connectivity to plant information, and simulation capability.

# 2
# CURRENT STATE TECHNOLOGY EVALUATION

The objectives of having an effective and efficient RISMC process will require developing an integrated suite of methods and tools to perform the individual tasks associated with performing the evaluations. This suite of tools will need to provide the capability to perform physics based analyses (for example, integrated neutron kinetics/computational fluid dynamics/thermal-hydraulics/and so on). It also will need to perform integrated risk evaluations to aggregate the assessments of multiple hazards, support advanced computational algorithms, alternate end states, uncertainty assessments and real time configuration risk management. Because of the level of maturity of the safety analysis and probabilistic risk assessment methods and tools currently in use, a logical first step is to conduct an assessment of the state-of-the-art of these tools.

## 2.1 Identification of Tools and Preliminary Analysis

The first step in the assessment of the current suite of tools employed to perform DSA and PRA for NPPs is to identify the spectrum of tools that are employed for these purposes. To accomplish this, an inventory and functional mapping of current DSA and PRA tools and application needs was performed. This task identified (to the greatest extent practicable given the time constraints for this project) a comprehensive listing of the tools used to perform NPP safety and risk analysis. For this effort, emphasis was placed on identifying those tools which were used throughout the industry to perform DSA and PRA (including configuration risk management) analyses. Due to the volume of information, the complete listing of the tools identified is provided in Appendix B. For ease of reference, the codes are listed in several tables grouped by function.

- Risk analysis codes (Table B-1)
- Computational fluid dynamics and thermal hydraulics codes (Table B-2)
- Configuration risk management codes (Table B-3)
- Fission product transport and dose assessment codes (Table B-4)
- Consequence analysis (PRA Level 3) codes (Table B-5)
- Structure analysis/risk assessment codes (Table B-6)
- Data management codes (Table B-7)

Additionally, analytical codes developed by EPRI are listed in Table B-8. For each code, a brief description is provided. Additionally, summary information also is provided that identified the code developer, identifies applicable reference material and provides an assessment of which level of a plant PRA the code can support.

Due to the limited time for performing the characterization of the current state of the art, the identified DSA and PRA codes were prioritized based on the use they receive in applications to analyses performed on commercial NPPs in the United States. The prioritization was used to determine which codes would receive a more in depth assessment based on the following logic.

- Priority 1: This priority includes codes that have widespread application for use in addressing plant operational or regulatory issues. DSA/PRA codes in this category were reviewed and characterized in detail and the results of the characterization are described in Chapters 3 and 4 of this report respectively.
- Priority 2: This priority includes codes that are applied to address plant operational or regulatory issues; however, their use is not as widespread by NPP operators as the codes classified as Priority 1. DSA/PRA codes in this category would be reviewed and characterized in this report if time and resources permitted.
- Priority 3: This priority includes codes that were identified to either not have widespread application to NPPs or were used in limited specialized applications. DSA/PRA codes in this category were not reviewed for this report.

The prioritizations were specified based on the review and opinions of several experts in the performance of DSA and PRA for NPPs (from the perspective of NPP operators). This resulted in the following prioritization assignments.

**Table 2-1**
**Prioritization of Deterministic Safety Analysis Codes**

| Priority 1 | Priority 2 | Priority 3 |
|---|---|---|
| CORETRAN | CENTS | COBRA |
| GOTHIC | CRAC2 | CONTAIN |
| LOFTRAN | FACTRAN | |
| MAAP | ICARE/CATHARE | |
| MACCS2 | MC3D | |
| MELCOR | | |
| RELAP | | |
| RETRAN | | |
| TRACE | | |
| VIPRE | | |

Note that in this listing of DSA codes, the TRAC codes are not listed because they have been integrated into and superseded by TRACE.

**Table 2-2**
**Prioritization of Probabilistic Safety Analysis Codes**

| Priority 1 | Priority 2 | Priority 3 |
|---|---|---|
| ATHEANA | PSAPAC | FIVE |
| CAFTA | RISKMAN | NUPRA/WinNUPRA |
| EOOS | ORAM/Sentinel | PRAQUANT |
| HRA Calculator | Safety Monitor | PRISIM |
| SAPHIRE | XINIX/FRANX | RBDA |
| PARAGON | UNCERT | RELEX |
| | | RiskSpectrum |
| | | SETS |
| | | Qrecover |

## 2.2 Methods and Code Assessments

For each code identified as Priority 1, a more detailed assessment of the code was performed. Each assessment provides a description of the code's functionality and the analyses for which the code can be used are listed. The code's computational structure and design philosophy are briefly described. The range of applicability of the code is described. Finally, any limitations and precautions are provided (typically in tabular format).

Chapter 3 provides an evaluation of the DSA codes identified as high priority. High priority PRA codes are evaluated in Chapter 4.

# 3
# DETERMINISTIC SAFETY ASSESSMENT METHODS AND CODES

The following sections provide descriptions of DSA programs in widespread use for analysis at commercial nuclear power plants. The following codes are included:

- CORETRAN
- GOTHIC
- LOFTRAN
- MAAP
- MACCS2
- MELCOR
- RAVE
- RELAP
- RETRAN
- TRACE
- VIPRE

## 3.1 CORETRAN

The CORETRAN code is a three dimensional (3D) core simulator aimed for both steady-state and transient calculations of LWR core models. It simulates the core from the inlet plenum to the outlet plenum. The code is based on an internal (explicit time scheme) coupling between the ARROTITA neutronics module, that solves the 2 neutron group (2-G) nodal diffusion equations and the VIPRE-02 thermal-hydraulics module, that employs a 6-equation two-fluid model. The code can be used for the analysis of reactivity initiated accidents as well as to study the 3D core dynamic transient response to system transients which generate thermal hydraulic boundary conditions (for example, temperature, pressure and coolant flow). However, as the neutronics modeling in RETRAN-3D is very similar, CORETRAN also can be used to initialize the core neutronic models for RETRAN-3D for coupled plant system analyses. The CORETRAN code requires a fuel type cross section library as an input. CORETRAN can be utilized to perform analysis of steady state depletion, operational transients and design basis transients (that is, core

design, core follow, thermal margin, rapid transients, and so on). CORETRAN also allows analysis of different types of fuel assemblies including simulation of partial length rods, water rods, channels, and so on as described by input variables.

The following are specific applications applicable to use of CORETRAN (see reference [9]):

- Plant support and fuel reload evaluation
  - Reload steady state core design
  - Reactor steady state core follow
  - Core thermal analysis
  - Mixed fuel core analysis
  - Xenon Events
- Reactor safety analysis and operational transients
- Reactivity excursions involving large power redistributions
  - LWR steam line break
  - PWR rod ejection
  - Anticipated transients without scram (ATWS)
  - Boron Dilution
  - Dropped rod
- Benchmarking simplified codes

Key CORETRAN features (See reference [9]):

- Advanced, efficient numerical scheme for steady state and transient analyses for both neutronic and thermal-hydraulics (T/H) modules
- Full LWR core and vessel steady state and transient simulation
- Variable nodalization scheme permits different grid structure in core and plena
- Time-dependent, analytic nodalization, two group diffusion theory
- Utilizes assembly discontinuity factors for enhanced accuracy
- Implicit core boundary treatment with baffle and reflector
- Steady state initialization capability, including rod bank position search and boron concentration search
- Two fluid models for thermal hydraulics
- Dynamic flow regime model (DFRM) for realistic, efficient flow regime simulation

CORETRAN can be utilized to determine the following information (See reference [9]):

- Loading pattern (Reload core design)
- Physics safety parameters
- Cross sections for kinetics calculation for input to system codes such as RETRAN-3D
- Core operating information
  - Rod positions and boron concentration for criticality
  - Information to assure shutdown margins
  - Reactivity requirements for xenon override
- Start-up predictions
  - Rod worth
  - Moderator or isothermal temperature coefficients
- In-core monitoring libraries (conversion factors)
  - Instrument signals to assembly average power
  - Assembly average power ratios
  - Ratio of maximum pin power to assembly average power
- Core follow (Track exposure of each assembly)
  - Power distribution changes
  - Rod position effects
- Local conditions in the area of failed fuel
- Conditions that can lead to dry-out
- Local sub channel sub-cooled boiling
- Condensing of local voids which move into colder regions
- Cross flow between adjacent fuel pins or assemblies
- Steady state or transient local pin power and associated heat transfer and fluid conditions

Reference [10] compares analyses of a Main Steam Line Break (MSLB) transient performed with CORETRAN/RETRAN-3D vs. the results of an identical transient analyzed with COSMOS-4/ SIMULATE-3 K both utilizing the same input cross section library. The MSLB is a highly asymmetric transient and is performed at hot zero power at the end of the fuel cycle. The resultant cool down causes a return to criticality and a return to power with a much skewed power distribution. Departure from nucleate boiling (DNB) and peak linear power generation are key fuel performance issues for this transient. Significant differences are shown to occur in the two analysis models due to the smaller moderator reactivity coefficient calculated in

CORETRAN. It is shown that this stems largely from differences in the cross-section formalism, that is, the manner in which feedback dependencies are modeled and interpolated for the cross-section sets.

Specifically, the CORETRAN cross-section model utilizes an inadequate treatment of coupled feedback effects, principally between boron density and moderator temperature, which impacts the MSLB transient predictions. As such, transient-specific cross-section libraries need to be produced for reliable MSLB analysis utilizing CORETRAN. The cross-section model for SIMULATE-3 K, on the other hand, is shown to be adequate for accurately capturing the coupled reactivity effects occurring during an MSLB. The paper points out that many other state-of-the-art advanced kinetics codes have cross-section formalisms similar to that of CORETRAN. Thus, effects of the type investigated in the paper need to be taken into account while developing methodologies for assessing neutronic predictions.

## 3.2 GOTHIC

The Generation of Thermal Hydraulic Information in Containments, Version 7.1 (GOTHIC7.1) is a computer program used for multi-phase, multi-component fluid flow. When applied to the nuclear industry, GOTHIC can be used for design, operating, safety and licensing analyses. GOTHIC is used to do the following:

- Analyze containment response to high energy line breaks within design basis analysis (DBA).
- Analyze equipment qualification.
- Perform room heat-up calculations.
- Investigate the deterioration or failure of engineered safeguard features.
- Analyze multi-phase flow in piping.
- Evaluate spent fuel performance.

GOTHIC7.1 results primarily are used to determine Level 1 and 2 success criteria and accident timing for probabilistic risk assessments (PRAs). They also are used for equipment qualification analyses, containment analyses in boiling water reactors (BWRs) and pressurized water reactors (PWRs), fission product large early release frequency (LERF) determinations, analyses to support plant modifications, generic plant issue assessments (for example, significance determinations) and other similar applications.

There are three codes included in the GOTHIC package: GOTHIC_P, GOTHIC_S and GOTHIC_G. GOTHIC_P is a preprocessor and postprocessor which is menu and pictorial driven. GOTHIC_P is used to develop a model and is where input parameters are stored. GOTHIC_P also is used to gather graphical output obtained in the analysis, as well as create graphs and input files. GOTHIC_P allows analysts to plot specific variables. GOTHIC_S is used to quantify the models created in GOTHIC_P. This program quantifies conservation of energy,

mass and momentum equations for multi-phase, multi-component flow. GOTHIC_G is a graphics program that allows analysts to create replicas of nuclear reactor auxiliary and containment buildings as well as retrieve results from graphics files.

### *3.2.1 Capabilities, Computational Structure and Documentation*

GOTHIC descends from the COBRA series of codes, which modeled multi-dimensional thermal-hydraulics in the reactor vessel. GOTHIC was developed by Numerical Applications, Inc. (NAI). The Electric Power Research Institute (EPRI) sponsors the development and maintenance of the program. NAI developed and maintains the program under a quality assurance program. This quality assurance program meets the requirements of U.S. 10CFR50 Appendix B and ASME NQA-1.

GOTHIC currently is used at more than 25 United States utilities for equipment qualification and safety analyses. Canada also uses GOTHIC for nuclear plant analysis as part of their Industry Standard Tool Set. GOTHIC is used for containment analysis at Mitsubishi Heavy Industries in Japan and KOPEC in Korea. The program also is used to support experimental programs at Research Institutes such as PSI (Switzerland) and CEA (France).

GOTHIC7.1 is used to analyze the spectrum of momentum transport terms embedded in multi-dimensional models. Optional models can be employed for turbulent mass and turbulent shear and energy diffusion. The following phenomena are modeled in GOTHIC7.1:

- Fluid flow
- Natural circulation
- Steam evaporation and condensation
- Boiling
- Conduction, convection, and radiation heat transfer
- Hydrogen dispersion
- Failure of containment building
- Ignition of hydrogen
- Radioactive isotope transport, decay and deposition
- Heat transfer between phases
- Heat transfer between surfaces and fluid
- Separate but interfacing field relative velocities

GOTHIC is suited for use on Unix workstations as well as PCs running on a Windows NT platform. The format of the input and output files is suited for plant engineers. GOTHIC models use a noding scheme which lets computational volumes be estimated as lumped parameters. Nodalization and large time steps allow GOTHIC to generate results quickly.

The GOTHIC7.1 program is documented in the GOTHIC7.1 User's Manual [11]. The user's manual describes how the three programs within the GOTHIC package can be used. The manual also gives instructions for creating an input model, running the code and gathering output.

### 3.2.2 Range of Applicability, Limitations and Precautions

In order to produce quality GOTHIC7.1 results for analyses, an understanding of the program's range of applicability and its limitations is necessary. Results generated from GOTHIC are important since they are used to determine PRA model success criteria. The GOTHIC User's Manual [11] lists the following information regarding limitations with the program.

**Table 3-1**
**GOTHIC7.1 Limitations**

| Affected Area | Comment |
|---|---|
| Compare Utility | The compare utility in GOTHIC allows analysts to substantiate and document that only intended and designated changes were made to the GOTHIC_P files. The compare files must be of the same GOTHIC version. In order to overcome this limitation it is necessary to load the older compare file and save the upgraded file. The two files then can be compared. |
| Linkage Utility | Users are able to link table data in GOTHIC to data in Excel or Access. The linkage utility only works in the Windows NT/95 version of GOTHIC. |

## 3.3 LOFTRAN

The LOFTRAN (Loss of Flow Transient) code originally was written in the early 1960's to simulate the response of the reactor coolant system to a loss of flow event. Originally, it combined all of the reactor coolant loops into one, requiring the use of other codes for transients that did not possess loop symmetry. In the mid 1970's, Westinghouse modified the code to simulate as many as four separate reactor coolant system primary loops.

Since its inception, the LOFTRAN code has been the non-LOCA transient code of choice for Westinghouse designed plants. It has been shown to have an excellent correlation with other similar reactor coolant system transient codes such as RETRAN (see reference [12]). More recently, Westinghouse has been utilizing RETRAN when three dimensional (3D) modeling capabilities are needed such as for application in their 3D comprehensive analysis code RAVE [13]. However, the current Design Control Document for the AP1000 utilized LOFTRAN as the non-LOCA transient analysis methodology. There are two probable reasons for this:

First, Westinghouse had obtained NRC approval for the use of the LOFTRAN code as modified for consideration of the passive features.

Second, the margins afforded by using 3-D methodology were not needed for the AP1000 analysis and there was a desire to avoid incurring licensing exposure and additional expense required to license RETRAN for analysis of the passive safety features.

The LOFTRAN code is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, and safety valves also are considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. Reactivity effects are simulated by moderator density coefficients that are functions of moderator density and boron concentration, boron coefficient and power coefficients to simulate the Doppler effect. The reactivity coefficients are generated by static, multidimensional neutronics codes. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, over-temperature  T, high and low pressure, low flow, and high pressurizer level. Control systems also are simulated, including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, also is modeled.

LOFTRAN is a versatile program suited to accident evaluation and control studies as well as parameter sizing. Westinghouse utilizes LOFTRAN to analyze essentially all of the non-LOCA transients except those that involve high positive reactivity insertion rates such as rod withdrawal from sub criticality and rod ejection. For those transients for which abnormal flux profiles are generated, the point kinetics state points (temperature, pressure power, flow and boron concentration) are chosen for the most severe point and analyzed with a multi-dimensional static neutron diffusion code to verify reactivity and to provide input to a thermal hydraulics code to determine the departure from nucleate boiling ration (DNBR). This approach yields defensible results as the transient is relatively slow and can be considered quasi-static (see reference [14]). This approach was verified by detailed 3D analysis using the RAVE methodology (see reference [13]).

LOFTRAN is used to estimate mass and energy releases for a spectrum of high energy secondary side pipe breaks for the containment and sub compartment temperature and pressure response.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits. The core limits represent the minimum value of DNBR as calculated for a typical or a thimble cell. (A typical cell is an array of four fuel pins; a thimble cell is an array of three fuel pins and one instrument thimble.)

Most of the transient analysis performed for the safety cases ignores the effects of heat transfer to and from the thick metal mass in the reactor coolant system. LOFTRAN has the ability to model the geometry, heat retention and the transfer of it to and from the metal mass, if this becomes significant in the progress of the transient. See reference [12].

For transient simulation of the advanced AP600 and AP1000 plant designs, the LOFTRAN code has been modified to allow the simulation of the passive residual heat removal (PRHR) heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. The LOFTTR2 Code is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate the impact of operator actions during a steam generator tube rupture (SGTR) event. For transient simulation of the AP600 and AP1000, the LOFTTR2 code also has been modified to allow the

simulation of the PRHR heat exchanger, core makeup tanks, and associated protection system actuation logic. The modifications are identical to those made to the LOFTRAN code.

## 3.4 MAAP

The Modular Accident Analysis Program Version 4 (MAAP4) is a computer code that simulates the response of light water reactor (LWR) power plants during severe accidents. Given a set of initiating events and operator actions, MAAP4 predicts the plant's response as the accident progresses. The code is used to do the following:

- Predict the timing of key events (for example, core uncovery, core damage, core relocation to the lower plenum, and vessel failure).
- Evaluate the influence of mitigative systems, including the impact of the timing of their operation.
- Evaluate the effect of operator actions.
- Predict the magnitude and timing of fission product releases.
- Investigate uncertainties in severe accident phenomena.

MAAP4 results primarily are used to determine Level 1 and 2 success criteria and accident timing for probabilistic risk assessments (PRAs). They are also used for equipment qualification analyses, fission product large early release frequency (LERF) determinations, integrated leak rate test evaluations, emergency planning and training, simulator verification, analyses to support plant modifications, generic plant issue assessments (for example, significance determinations) and other similar applications.

MAAP4 is an integral code. It treats the full spectrum of important phenomena that could occur during an accident, simultaneously modeling those that relate to the thermal hydraulics and to the fission products. It also simultaneously models the primary system and the containment and reactor/auxiliary building.

There are parallel versions of MAAP4 that support boiling water reactors (BWRs) and pressurized water reactors (PWRs). These two versions contain the same core model, containment and reactor/auxiliary building model, fission product model, and input and output schemes. They have distinct primary system models and engineered safeguards models. The code is applicable to both current and advanced LWR designs, with models that represent the passive features of the latter.

A new version of the code, MAAP5, has been completed and is under limited use by several MAAP users. The major advancements include more detailed thermal-hydraulic modeling for the PWR primary system along with neutronics modeling.

### *3.4.1 MAAP Development History and MAAP Users Group*

MAAP was originally developed for the Industry Degraded Core Rulemaking (IDCOR) program in the early 1980s by Fauske & Associates, LLC (FAI). At the completion of IDCOR, ownership

of MAAP was transferred to the Electric Power Research Institute (EPRI), which was charged with maintaining and improving the code. The code has been developed and is maintained under a quality assurance program, which is in compliance with U.S. 10CFR50 Appendix B and ISO 9001 quality assurance requirements.

### *3.4.2 Phenomena Modeled in MAAP4*

MAAP4 treats the spectrum of physical processes that could occur during an accident. Level 1 PRA phenomena include the following:

- Gas and water flow
- Natural circulation
- Steam evaporation and condensation
- Boiling
- Critical flow
- Conduction, convection, and radiation heat transfer
- Countercurrent flow

Level 2 PRA phenomena include the following:

- Cladding oxidation and hydrogen evolution
- Core material eutectic formation
- Core relocation
- Lower head core debris dynamics
- Failure of vessel penetrations and/or the lower head
- Debris entrainment
- Debris-concrete interactions
- Ignition of combustible gases
- pH and iodine chemistry in containment
- Fission product release, transport, and deposition

### 3.4.3 Computational Structure and Design Philosophy of MAAP4

The MAAP4 code is written primarily in Fortran and can be run on a variety of computer platforms, most commonly PCs. The format of the input and output files is tailored to plant engineers. The equations in MAAP4 are essentially lumped parameter, nonlinear, ordinary differential equations in time. The models in MAAP4 have been designed so that the code is fast running. This is a hallmark of MAAP. The primary means of achieving this objective are the use of quasi-steady modeling wherever appropriate, relatively coarse nodalization, and the largest possible time step consistent with the level of detail desired. The result is that the code execution time is generally several orders of magnitude faster than problem time on a typical PC and considerably faster than most comparable codes.

### 3.4.4 MAAP4 Documentation

The MAAP4 code is documented in the MAAP4 User's Manual [15], the user's guides (that is, the six sample parameter files), and the MAAP4 transmittal documents [16, 17].

The user's guides contain detailed descriptions and default values and ranges of the input parameters included in the parameter file. There are BWR guides for Mark I, Mark II, and Mark III containments and PWR guides for Westinghouse large dry and ice condenser plants with U-tube steam generators and for B&W plants with One Through Steam Generators (OTSGs). The guides are essentially sample parameter files and can be used as templates for plant-specific parameter files.

### 3.4.5 Range of Applicability, Limitations and Precautions

One criterion for producing high quality MAAP4 analyses is an understanding of the code's range of applicability and its limitations. Of particular importance is the applicability of the code for generating results that can be used to determine success criteria and HRA timing. The MAAP Users Group has published the following information regarding specific limitations with the code.

**Table 3-2**
**MAAP4 Limitations**

| Accident Definition | Comment |
|---|---|
| Double Ended Guillotine Cold Leg Break | Since the accident causes the flow to reverse initially, do not use MAAP until reflood is complete. Use DBA codes during this interval. After reflood MAAP will track the accident sequences. |
| Double Ended Hot Leg Rupture | Flow in the core does not reverse and MAAP can be used. |
| Large Break Cold Leg LOCA but Less Than a DECL Break (Leak-Before-Break) | If the flow within the core is not reversed, MAAP will calculate the appropriate heatup and potential shutdown of the nuclear reaction; benchmark with LOFT FP-2 demonstrates the code capabilities. |
| Medium LOCA | Since the flow does not reverse within the core, MAAP can be used for such success criteria. |
| Small Break LOCA | MAAP treats the behavior under small break LOCAs quite well. This is evidenced by the successful benchmark with the TMI-2 accident behavior. Also, the MAAP model has been successfully benchmarked with the Prairie Island steam generator tube rupture. |
| Loss of Heat Sink Accidents | MAAP represents the behavior of the core under these conditions quite well. This is best evidenced by the benchmarks with the Davis-Besse loss-of-feedwater event (PWR) and the Oyster Creek loss-of-feedwater event (BWR). |
| Main Steam Line Break | This is a rare initiating event for severe core damage, but the MAAP model has been benchmarked with the Westinghouse MB-2 experiments for steam generator response to loss-of-feedwater, MSLB, and so on. |

## 3.5 MACCS2

The MELCOR Accident Consequence Code System Version 1.13.1 (MACCS2) is a computer code designed to evaluate dose consequences as a result of severe accidents at nuclear power plants. MACCS2 also considers the removal of particulate nuclides from the plume via wet disposition. The code is used to do the following:

- Perform sensitivity analyses.
- Evaluate the fifty-year Total Effective Dose Equivalent (TEDE).
- Perform deterministic consequence analyses.
- Evaluate complimentary cumulative distribution functions (CCDFs).
- Perform cost/benefit analyses.

MACCS2 results primarily are used to determine Level 3 success criteria for probabilistic risk assessments (PRAs). They also are used for emergency planning and training, analyses to support plant modifications, and other similar applications.

MACCS is a unique code in that it is capable of modeling short-term and long-term mitigative actions, and the economic costs associated with them. MACCS2 also models deterministic and stochastic health effects in conjunction with mitigative actions.

Modeling with MACCS2 is divided into three modules: ATMOS, EARLY and CHRONC. ATMOS employs a Gaussian plume model with Pasquil-Guifford dispersion parameter. It evaluates atmospheric transport, the scattering of material and its displacement from the air. EARLY is used to model the consequences of accidents to the NPP's surrounding areas during emergency action periods. CHRONC evaluates the long-term impact in the time frame following the emergency action duration. The ATMOS, EARLY and CHRONC modules are evaluated in successive order.

MACCS2 version 1.13.1 addresses coding errors from previous versions of the program. This new version corrected the coding errors with regard to generating intermediate-phase results as well as implementing the dose and dose rate reduction factor (DDREF), which is used to estimate cancer risks.

### 3.5.1 Capabilities, Computational Structure and Documentation

The first version of MACCS, version 1.4, was developed by Sandia National Laboratories (SNL) in 1987. SNL is charged with the development and maintenance of MACCS2. SNL receives funding for MACCS2 development from the U.S. Department of Energy (DOE), as well as the U.S. Nuclear Regulatory Commission (NRC). Supplemental DOE work is supported by the Los Alamos National Laboratory (LANL). MACCS2 currently is used at nuclear power plants across the United States.

MACCS2 evaluates a host of processes that could occur during and after a severe accident. Level 3 PRA phenomena include the following:

- Source term specification
- Weather data
- Risk dominant plume
- Plume dimensions
- Atmospheric transport
- Plume depletion by radioactive decay, wet deposition and dry deposition
- Air and ground radionuclide concentrations

The MACCS2 code is written in FORTRAN 77 and 90 and can be run on Pentium PCs. Equations employed in MACCS2 are simple mathematical equations. The code is fast running and generates results quickly. Quantiles are estimated using a log-linear interpolation. MACCS2 calculations are divided into modules and phases. The code is comprised of simple models with analytical solutions. Performing calculations with MACCS2 involves three phases: input

handling and validation, phenomenological modeling and output handling. The basis for the phenomenological modeling is empirical data. The results are usually analytical and in text file format.

The MACCS2 code is documented in the MACCS2 Application Guide [18]. The application guide provides direction on the use of MACCS2 for safety basis analysis, as well as building input files for evaluation. NUREG/CR-6613 [19] describes the code as well. It also guides users on preparing input files and interpreting results.

### *3.5.2 Range of Applicability, Limitations and Precautions*

In order to produce quality MACCS2 results for analysis, an understanding of the code's range of applicability and its limitations is necessary. Of particular importance is the applicability of the code for generating results that can be used to determine success criteria for Level 3 PRAs. The MACCS2 Users Group has published the following information regarding specific limitations with the code.

**Table 3-3**
**MACCS2 Limitations**

| Accident Definition | Comment |
|---|---|
| Temporal Regime | The use is best suited for "short" duration plumes, ranging from approximately several minutes to 10 hours. |
| Spatial regime | The code does not model dispersion close to the source (less than 100 meters from the source), especially where the influence of structures or other obstacles is still significant. Similarly, the MACCS2 class of codes (that is, Gaussian models) should be applied with caution for significant distances (that is, greater than 50 miles from the source), especially if meteorological conditions are likely to be different from those at the source of the release. Long-range projections of dose conditions are better calculated with mesoscale, regional models that are able to account for multiple weather observations. |
| Terrain Variability | Gaussian models are inherently flat-earth models, and perform best over regions of transport where there is minimal variation in terrain |
| Thermal buoyancy | In plumes arising from fire-related source terms, the user should exercise caution with the models such as MACCS2 that use the Briggs algorithm. The Briggs approach for accounting for sensible energy in a plume is valid for "open-field" releases (that is, releases that are not impacted by buildings and other obstacles), or if the code is used in combination with building wake effects. |

## 3.6 MELCOR

The Methods for Estimation of Leakages and Consequences of Releases version 2.1 (MELCOR2.1) is a computer code that is used to model the response of light water reactor (LWR) plants during severe accidents. MELCOR2.1 consists of various modules. These combined modules are used to model the important systems of nuclear power plants. Given a set of initiating events, MELCOR2.1 evaluates the plant's response as the accident progresses in both boiling water reactors (BWRs) and pressurized water reactors (PWRs). The code is used to do the following:

- Predict the timing of key events (for example, core uncovery, core damage, core relocation to the lower plenum, and vessel failure).
- Predict the magnitude and timing of fission product releases.
- Analyze the influence of mitigative systems.
- Evaluate the Leak Path Factor (LPF) for postulated accident conditions.
- Analyze uncertainties and sensitivities in severe accident phenomena.
- Examine design basis accidents for advanced LWR plant applications (for example, ESBWR, EPR, and APWR).

MELCOR2.1 results are used to determine Level 1 and 2 success criteria and accident timing for probabilistic risk assessments (PRAs) in the nuclear industry. They also are used for equipment qualification analyses, fission product large early release frequency (LERF) determinations, integrated leak rate test evaluations, emergency planning and training, analyses to support plant modifications and initial plant design, generic plant issue assessments (for example, significance determinations of plant events/reported incidents) and other similar applications. MELCOR also models postulated releases due to fire and seismic events.

MELCOR2.1 treats the full spectrum of important phenomena that could occur during an accident, simultaneously modeling those that relate to the thermal hydraulics and to the fission products. It also simultaneously models the primary system and the containment and reactor/auxiliary building.

Several modules with distinct purposes are available in MELCOR. These modules make modeling in MELCOR flexible. The degree of detail used when building the model is optional as no specific system nodalization is provided in MELCOR. One model is used to represent the core in both BWRs and PWRs. The user modifies the code to make the core model, containment model and reactor/auxiliary building model, fission product model, and input and output schemes plant-specific. MELCOR is sufficiently flexible to be used for nonreactor issues. For example, designs employed in Eastern European reactors currently utilize MELCOR for source term calculations.

### 3.6.1 MELCOR Development History and the MELCOR Users Group

MELCOR was developed by the Sandia National Laboratories (SNL) and is funded by the U.S. Nuclear Regulatory Commission (NRC), the U.S. Department of Energy (DOE) and the International Cooperative Severe Accident Research Program (CSARP). The Sandia National Laboratories remains responsible for the development and maintenance of MELCOR. MELCOR has been developed and is maintained under a quality assurance program, which is in compliance with U.S. 10CFR830.

The MELCOR Users' Group meets annually to discuss capabilities of the code, new features, utilizing MELCOR to simulate various reactor and nonreactor issues and any identified problems with the coding.

### 3.6.2 Phenomena Modeled in MELCOR2.1

MELCOR treats the spectrum of physical processes that could occur during an accident. Level 1 PRA phenomena include the following:

- Gas and water flow
- Critical flow
- Countercurrent flow
- Natural circulation
- Conduction, convection, and radiation heat transfer
- Steam evaporation and condensation
- Boiling

Level 2 PRA phenomena include the following:

- Cladding oxidation and hydrogen evolution
- Ignition of combustible gases
- Chemistry in containment
- Fission product release, transport, and deposition
- Core material eutectic formation
- Core relocation
- Lower headcore debris dynamics
- Failure of vessel penetrations and/or the lower head
- Debris-concrete interactions
- Debris entrainment

### 3.6.3 Computational Structure, Design Philosophy, and Documentation

The MELCOR code is written primarily in FORTRAN and can be run on a UNIX workstation as well as PCs. MELCOR employs lumped parameter, linear equations to estimate the airborne source term. The models in MELCOR are designed so that the code is fast running. The result is that the code execution time is generally several orders of magnitude faster than problem time on a typical PC.

The MELCOR code is documented in the MELCOR Computer Code Manuals [20], the MELCOR Guidance Report [212], and the MELCOR Gap Analysis Report [22]. The MELCOR Computer Code Manuals contain detailed descriptions of the input guidelines and instructions for each module in MELCOR. The guidance report provides direction as to how analysts can use MELCOR for safety analysis. The gap analysis report describes the software quality assurance program used to develop and maintain MELCOR.

### 3.6.4 Range of Applicability, Limitations and Precautions

One criterion for producing high quality MELCOR analyses is an understanding of the code's range of applicability and its limitations. Of particular significance is the applicability of the code for generating results that can be used to determine success criteria. The MELCOR Guidance Report has noted the following limitations within the code.

**Table 3-4**
**MELCOR2.1 Limitations**

| Affected Area | Comment |
|---|---|
| Gas or aerosol mixing/transport | The control volume method employed in MELCOR utilizes lumped parameter models. This approach does not model mutli-dimensional issues, such as the formation of gases within a room. To adequately model this phenomenon, the room will need to be separated into more volumes. This approach can be augmented with the use of computational fluid dynamics (CFD) code results. |
| Fluid Flow | MELCOR does not properly evaluate processes such as the flow of fluid in pining and ducting systems. As a result when investigating ventilation systems, vent ducts should not be modeled to credit deposition of aerosolized masses. If the aforementioned is included, it will increase the evaluation time and have no significant impact on results. |

## 3.7 RAVE

The RAVE methodology was developed by Westinghouse in order to free up a large amount of margin that is consumed because of the overly conservative assumptions made in previous safety analyses. RAVE builds on the 3-D neutronics and thermal hydraulics capability of SPNOVA and VIPRE coupled with the systems modeling features of RETRAN. All three codes are run in parallel trading information between each other. No changes were made to the codes themselves; RAVE manages the data such that the codes can be run in parallel [23].

The SPNOVA code performs the steady state and 3D transient kinetics. VIPRE provides local temperature and density effects for the SPNOVA determination of Doppler and moderator reactivity feedback. VIPRE also is used to calculate the local heat flux that is used in the RETRAN model. RETRAN is used to model the Reactor Coolant System (RCS) response. The reactor vessel, RCS loops, pressurizer, reactor coolant pump behavior and steam generators are all specifically modeled in nodal form. Also modeled are the engineered safety features, reactor trip signals as well as the control functions such as pressurizer heaters and spray, feed water flow and turbine controls, and so on. Just as in the traditional safety analysis report accident analyses, the control systems are assumed not to operate unless operation would aggravate the consequences of the accident. The neutron kinetics and fuel rod heat transfer features of RETRAN are not used. VIPRE is used to calculate the local time variant heat flux as an input to RETRAN. In turn, RETRAN provides the core inlet flow and temperature as well as core exit pressure for the VIPRE thermal hydraulic calculations. VIPRE also calculates the hot rod minimum DNBR as a function of time as well as the hot rod time dependent fuel and clad temperatures. In addition to nodal treatment of the RCS loops and components, the fuel is nodalized axially and radially to better predict the thermal and reactivity feedback effects.

Previously, ultra conservative assumptions were made in performing NPP safety analyses. Often these assumptions were contradictory; a common example is combining beginning of cycle properties with end of cycle properties so that the analysis can be said to bound all possible cases. Margin is freed up because of two significant features of RAVE:

- The codes are run in parallel rather than sequentially as was done in the past. This allows a much more realistic modeling of the interaction and feedback between the neutronic effects and the thermal hydraulic behavior.
- The 3-D modeling of the neutronics and the thermal hydraulics allows much more realistic prediction of the local effects such as cross flow and local reactivity feedback.

Take as an example, the Loss of Flow transient. The traditional Westinghouse Chapter 15 analysis shows that the minimum DNBR occurs at some time after the rods begin to fall into the core. Previous analysis would have generated trip reactivity as a function of rod position assuming the axial flux profile peaked towards the bottom of the core. This would minimize the amount of negative reactivity inserted at the beginning of the transient thus maintaining the power level at a higher value when the minimum DNBR occurs thus minimizing the predicted value of DNBR. Conversely, when the DNBR was calculated with the previous methodology, the axial neutron flux was assumed to be peaked towards the top of the core in order to cause the power level to be higher at positions of higher coolant enthalpy thus further under-predicting the DNBR. These results are clearly contradictory and cause a substantial loss of predicted margin. With RAVE, the analyst can allow the code to predict the power distribution and allow local effects to govern.

Many other advantages are gained by the parallel coupling and 3-D nature of the modeling. However many of the traditional conservatisms are retained such as:

- Most limiting time in cycle
- Initial parameters assumed to be at the Technical Specification minimum or maximum (whichever is more limiting)
- Most reactive control rod is stuck out or fails to trip
- Independent single failure of a safety component
- Conservative rod insertion times for reactor trip
- Conservative delay times for engineered safety functions and trip times
- "First out" protection channel signal is ignored
- Minimum shutdown margin at any time in life
- Conservative reactor coolant pump coast down time

Westinghouse also states in Reference [24] that the reload safety methodology "Bounding Approach" developed in WCAP-9272-P-A/WCAP-9273-NP-A is supported by the RAVE approach. Reference [24] below provides further insight as to the interaction of the codes and provides descriptions of code capabilities.

## 3.8 RELAP

The Reactor Excursion and Leak Analysis Program (RELAP) is a tool for analyzing small-break LOCAs and system transients in PWRs or BWRs. It has the capability to model thermal-hydraulic phenomena in one dimensional (1D) volumes. While this code still enjoys widespread use in the nuclear community, active maintenance will be phased out in the next few years as usage of TRACE grows.

RELAP can be used in two configurations:

1. RELAP5 systems thermal-hydraulic software uses multi-dimensional thermal-hydraulics, heat transfer, generic component, and control systems models to describe the behavior of complex systems that operate with water and other noncondensable gases under single phase and two phase flow conditions. Typical system models can run significantly faster than real time on currently available PC's and engineering workstations.
2. RELAP/SCDAPSIM nuclear systems software uses RELAP in combination with specialized models to treat the behavior of the reactor system during accident conditions including those transients that may result in the failure of the core and reactor coolant system.

The development of the RELAP family of codes was started more than 3 decades ago by the U.S. government. Although RELAP5 originally was developed to support the analysis of postulated accidents in commercial nuclear power plants in the United States, different versions of the code have been widely distributed around the world and now are used to support a wide range of activities. The code has been used (a) to support basic research on two phase thermal-

hydraulics, (b) to design small and large scale thermal-hydraulic experimental facilities, research reactors, and commercial power plants, and (c) to assess the safety of nuclear power plants. RELAP/MOD2 was one of the most widely used versions of the software and is still used to support the regulation of commercial power plants in the United States, Europe, and Asia. RELAP/MOD3, which was initially released in the late 1980's, is the most advanced major version of the code and is still under active development in the United States by the U.S. Department of Energy and Nuclear Regulatory Commission (USNRC). RELAP/MOD3.2 is currently the latest publicly available version of the code [25].

The development of SCDAP/RELAP, which incorporates RELAP5 system models, was started in 1981 by the USNRC as a result of the severe accident at Three Mile Island Unit 2. This code, which has arguably the most advanced accident analysis models in the world, also has been used to support severe accident research programs as well as the assessment of the safety of nuclear power plants.

RELAP and SCDAP/RELAP use multidimensional thermal-hydraulic, heat transfer, generic and special components, control systems, and other models to describe the behavior of complex fluid-filled systems under single and two-phase flow conditions. The hydrodynamic models track the flow of liquid, vapor, and non-condensable gases including air, hydrogen, and nitrogen. The heat transfer models describe (a) 1D/2D heat conduction in system structures and (b) convective and radiative heat transfer between the structures and the fluid. The generic component models include valves, separators, dryers, pumps, electric heaters, turbines, and accumulators. Control system models include arithmetic functions, integrating and differentiating functions, proportional-integral, lead, and lead-lag controllers, and Boolean trip logic. Special component models in SCDAP/RELAP, developed for the analysis of nuclear reactors, include fuel element, control rod/blade, and other core structure models, debris bed models, and general models for porous structures.

Both codes have also been widely used by regulatory and research organizations around the world to support international standard problem exercises and experimental programs so the impact of user experience, the ability of the codes to predict thermal-hydraulic and severe accident phenomena, and applicability of the codes to prototypical plant transient data have been extremely well characterized. Although plant data for accident conditions is limited, these codes also have been widely used to assess the performance of NPPs under design basis and severe accident conditions. In particular, SCDAP/RELAP was used extensively by the US Department of Energy and many international organizations to support the assessment of the TMI-2 accident.

The development of RELAP/SCDAPSIM was started in 1999 under the sponsorship of an international consortium as part of the SCDAP Development and Training Program (SDTP) using models developed initially for RELAP and SCDAP/RELAP by the USNRC. The first version, RELAP/SCDAPSIM/MOD3.1, was released in 1997. At the time of the release, this code was the most advanced systems thermal-hydraulics package available for the PC using the WINDOWS 95/98/NT and LINUX operating systems. The code is based on the models used in the SCDAP/RELAP/MOD3.1e code released by the USNRC.

SCDAP/RELAP/MOD3.1e includes the most advanced models for the treatment of the early stages of a severe accident in a commercial nuclear power plant. The code also is capable of providing realistic bounding estimates of the later stages of a severe accident involving the

formation of large molten pools, debris beds, and attack of the reactor vessel. The systems thermal-hydraulic models in the code are significantly improved over previous versions of RELAP or SCDAP/RELAP5. Transients involving limited amounts of non-condensable gases with pressures well above atmospheric pressures can be evaluated. Transients with higher concentrations of non-condensable gases and pressures near atmospheric pressures also can be analyzed but may require the user to modify time steps employed for the calculations.

The first experimental version of RELAP/SCDAPSIM/MOD3.2 was first released in November of 1998. The code used models taken from the SCDAP/RELAP5/MOD3.2 code. SCDAP/RELAP/MOD3.2 had a number of significant modeling improvements, in particular, the treatment of the later stages of a severe accident. The first production version of RELAP/SCDAPSIM/MOD3.2, released in January 1999, included improvements in the numerics and programming implementation of the SCDAP/RELAP/MOD3.2 models, resulting in noticeable improvements in the speed and reliability of RELAP/SCDAPSIM/MOD3.2 relative to SCDAP/RELAP/MOD3.2 (and previous versions of RELAP/SCDAPSIM, RELAP5, and SCDAP/RELAP [26].

RELAP/SCDAPSIM utilizes SCDAP/RELAP thermal-hydraulic, fuel behavior, and other models developed by the USNRC, in combination with SDTP-developed models to describe the behavior of the reactor system.

The RELAP5-based models calculate the overall system response including the transport of materials through the system, control system behavior, reactor kinetics, and heat transfer between the system structures and the fluids. The thermal-hydraulic models utilize a multi-dimensional, two-fluid, non-equilibrium approach to describe the material transport including the effects of single phase and two phase convective heat transfer. Radiation heat transfer between the structures and the fluid is described using detailed network models. Heat conduction within the structures is described using 1D/2D models. The models also describe the transport of aerosols, fission products, hydrogen, air, nitrogen, and other non-condensable gases [25].

The SCDAP-based models calculate the heat up and damage progression in the core and surrounding structures. When applied to analysis of severe accidents, the SCADAP-based models describe the (a) heating, deformation, oxidation, and melting of fuel rods, control rods/blades, and other representative vessel structures and (b) formation, heating, and melting of debris. The heating, melting, oxidation, and changes in core and vessel structures are described using representative 2D component models. Physical processes predicted include (a) heat conduction within the structures, (b) fuel rod ballooning and rupture, (c) oxidation, (d) material interactions between the fuel, cladding, and structural and control materials, (e) fission product release, (f) spalling of protective oxide films, (g) relocation and freezing of molten films, rivulets, and droplets, and (h) fragmentation and collapse of the structures during reactor vessel reflood [25].

The behavior of debris beds, molten pools, and associated structures are described using a combination of lumped parameter and detailed 1D/2D finite element models. Physical processes predicted include heat conduction with the debris and embedded or adjacent structures, molten pool formation and growth, natural circulation heat transfer between the molten pool and boundary, molten pool crust thinning and failure, relocation of the molten material, and the failure of the structures due to thermal and creep rupture mechanisms [25].

RELAP/SCDAPSIM fission product and aerosol deposition models are based on the TRAP-MELT models that were originally modified for use in early versions of SCDAP/RELAP. Physical processes described by the models include evaporation and condensation, chemisorption, agglomeration, and deposition.

## 3.9 RETRAN

EPRI began an extensive program to develop RETRAN in 1975. The result, released in 1978, is a variable node reactor coolant system simulation code that can incorporate many features of light water reactor designs.

RETRAN-0l featured:

- A one-dimensional, homogeneous equilibrium mixture (HEM) thermal-hydraulic representation of the reactor coolant system (RCS)
- A point neutron kinetics model for the reactor core
- Auxiliary component models, including a non-equilibrium pressurizer model and a temperature transport delay model for pipe like regions of the RCS
- A versatile control system model that allowed construction of customized control and protection system representations using “control blocks,” or numerical representations of various analogue modules such as summers, amplifiers and filters
- A steady state initialization technique

At the time of the RETRAN-0l code release, a number of theoretical limitations to the code were known and documented. The subsequent RETRAN-02 code development effort was initiated to remove some of these limitations and to extend the capabilities of the code, particularly in the areas of modeling Boiling Water Reactor plants and transients associated with them. Additional enhancements were made to support modeling of small break loss of coolant accidents, anticipated transients without scram (ATWS) and certain balance of plant features, such as turbines.

To address these needs, a number of the RETRAN-0l models were revised and/or extended. Revisions included:

- An improved solution technique for the non-equilibrium pressurizer model
- A modified critical flow solution
- An equation of state for water that is valid over the range 0.1 psia to 6000 psia.
- A revised momentum mixing calculation (primary for modeling BWR jet pumps)

In addition, RETRAN-02 includes the following additional models:

- Dynamic and algebraic slip models for two-phase flow
- One dimensional space-time neutron kinetics model
- A set of two-phase natural convection heat transfer correlations
- An iterative solution scheme for the fluid field equations
- A turbine model and a condensing heat transfer model for balance of plant analyses
- Local conditions heat transfer model (important for ATWS and other severe loss of inventory conditions).
- Vector momentum representation of the fluid
- An auxiliary model (profile fit) to compute void fraction for void reactivity feedback (primarily for BWR5 designs)
- Thermo/physical properties and a forced convection heat transfer correlation for supercritical water
- Steam separator efficiency model (primarily for BWRs)

RETRAN-02 is a versatile and reliable computer program for use in best-estimate transient thermal-hydraulic analysis of light water reactor systems. It is based on a one-dimensional homogeneous equilibrium mixture model with an optional phasic slip formulation based on either a drift flux model or a phasic velocity difference differential equation. RETRAN-02 contains both point reactor and one-dimensional kinetics models and component models for reactor control systems, pressurizers, and separators. It also has flexible heat conduction and heat transfer models that allow modeling of both reactor cores and steam generators. A unique capability of RETRAN-02 is its steady-state initialize feature, which aids users in establishing the desired initial state for transient simulations. RETRAN-02 provides analysis capabilities for (1) BWR and PWR transients, (2) small break loss of coolant accidents, and (3) anticipated transients without scram. It is used to support plant licensing, operational support, and training issues.

### 3.9.1 RETRAN-3D

The RETRAN-3D computer program evolved from the continued development of the RETRAN codes. It represents the latest in modeling capabilities and features which include the ability to

reproduce RETRAN-02 results. Like its predecessor, RETRAN-02, RETRAN-3D was developed to perform licensing and best-estimate transient thermal-hydraulic analyses of light water reactors. It is maintained under a Quality Assurance program in compliance with 10CFR50 Appendix B. The RETRAN-3D code development was sponsored by EPRI. [27, 28]

RETRAN-3D is a well accepted transient thermal-hydraulic analysis code designed for use in best-estimate evaluation of light water reactor systems. It is based on the one-dimensional homogeneous equilibrium model (HEM). Two additional modeling options allow for increasing levels of improved modeling capability for two-phase flow conditions. One makes use of a slip model (dynamic or algebraic) to augment the HEM equations and eliminate the equal velocity assumption when two-phase conditions exist. The other uses both a slip model and the vapor continuity equation to augment the HEM balance equations. This eliminates the equal velocity and equal temperature assumptions of the HEM model. A gas continuity equation can be added to any of the three options described above to allow non-condensable gas flow to be modeled.

When using RETRAN-3D, heat generation in a nuclear reactor core can be modeled using three-dimensional, one-dimensional, or point reactor kinetic models. Component and auxiliary models allow for complete modeling capability for the nuclear steam supply system including controls. Many new models have been added to RETRAN-3D and many RETRAN-02 models have been revised to improve the accuracy of their results.

RETRAN-3D retains the analysis capabilities of RETRAN-02 in that it can be applied to (1) BWR and PWR operational transients, (2) small break loss-of-coolant accidents, and (3) anticipated transients without scram. RETRAN-3D has much improved capabilities for the last two transient types and also has model extensions designed to provide analysis capabilities for (1) long-term transients, (2) transients that possess thermodynamic nonequilibrium phenomena, (3) PWR mid-loop operation with noncondensable gas present, (4) transients where three-dimensional power shapes and reactivity feedback effects are important, and (5) BWR stability events.

RETRAN-3D has been approved by the United States Nuclear Regulatory Commission for use in performing licensing calculations and is used by a large number of domestic and foreign electric utilities and research organizations. RETRAN-3D MOD004.3 is the current version, and is operational on PCs and UNIX-based workstations. A list of trouble reports and their status can be obtained from the RETRAN Trouble Report page. A useful application of a of RETRAN for utility applications is provided in Reference [29] (available from the NRC Public Document Room under ADAMS Accession Number ML042590169).

## 3.10 TRACE

The TRAC/RELAP Advanced Computational Engine is a modernized thermal-hydraulics code designed to consolidate the capabilities of the NRC's 3 legacy safety codes - TRAC-P (PWR), TRAC-B (BWR) and RELAP. It is able to analyze both large and small break LOCAs and system transients in both PWRs and BWRs. The capability also exists to model thermal hydraulic phenomena in both one and three dimensions. As a result, TRACE has become the NRC's primary thermal-hydraulics analysis tool. A comprehensive validation matrix including separate and integral effect tests is being identified for the overall code assessment and validation in different areas.

As part of the international CAMP-Program of the USNRC, the best-estimate code system TRACE coupled with the Purdue Advanced Reactor Core Simulator (PARCS) is being qualified by different institutions worldwide. The coupling of TRACE and PARCS is intended to take into account the interaction of the plant dynamic thermal-hydraulic performance and the neutron kinetic core behavior. In this way, the TRACE-application range fully covers that of the classical safety analysis codes previously employed by the USNRC (for example, RELAP5, TRAC-P, TRAC-B and RAMONA) and go beyond their individual capabilities. In addition, TRACE is able to simulate the behavior of nuclear reactors with working fluids other than water, for example, helium, carbon dioxide, liquid metal (lead-bismuth), and so on. A general user interface makes possible the coupling of TRACE with different program modules devoted to areas such as 3D-kinetics (PARCS), and containment thermal hydraulics (CONTAIN).

For post-processing, the Program AcGrace is available to present multiple and complex evaluations of the calculation’s results. A Symbolic Nuclear Analysis Package, SNAP, is a graphical user interface with pre -and post- processing capabilities that assist the user in the development of RELAP5 input decks and in running the code. The post-processor includes a "simulator-like" visualization and run-time control capabilities. SNAP is being developed to assist the preparation of input deck development for most of the US NRC code systems like RELAP5, TRAC, TRACE, VICTORIA, and so on. A special feature is the automatic conversion capability of both TRAC and RELAP5 input decks into TRACE-input decks.

In the present stage of development, the RELAP5/PARCS code system is able to simulate both PWR and BWR NPPs with a 3D-neutronic model coupled to 1D-RELAP5 thermal hydraulics (with RELAP kinetics turned off). A new version of PARCS is capable to additionally simulate hexagonal fuel assemblies (for VVER designed plants). This coupled code system makes use of an internal integration scheme where the system solution and core thermal hydraulics are obtained by RELAP5 and only the spatial kinetics solution is obtained by PARCS. In this approach PARCS uses the thermal hydraulic parameters predicted by RELAP5 to update the neutron cross-sections (feedback) while RELAP5 takes the thermal power calculated by PARCS to solve the core heat conduction problem. The temporal coupling is explicit. A general interface (GI) is used to manage the passing of variables according to the mapping between the thermal hydraulic and neutronic model from RELAP5 to PARCS and vice versa. Both the GI and PARCS code are executed as separate processes that communicate to each other via the message-passing protocol in the Parallel Virtual Machine (PVM).

## 3.11 VIPRE

The Versatile Internals and Component Program for Reactors (VIPRE) code is used for analysis of nuclear reactor thermal-hydraulics. It was designed to evaluate nuclear reactor core safety limits including the minimum departure from nucleate boiling ratio (MDNBR), critical power ratio (CPR), fuel and clad temperatures, and coolant state during both normal operation and under assumed accident conditions. VIPRE-01 initially was developed by Battelle Pacific Northwest Laboratories under the sponsorship of the Electric Power Research Institute (EPRI) and submitted to the NRC for generic review in 1984 (References [30, 31]). Currently the VIPRE User Group (VUG) is managed by CSA for EPRI with VUG members funding ongoing code maintenance and development activities.

There exist two basic version of VIPRE. VIPRE-01 is a general-purpose analysis that evaluates NPP thermal-hydraulic performance under normal operating conditions, operational transients, and events of moderate severity. The code requires input of boundary conditions that describe the coolant entering the core, core power generation, and the dimensional and material properties of the nuclear fuel. The boundary conditions for the coolant entering the core include the inlet flow rate, enthalpy and pressure or, alternatively, the pressure, inlet enthalpy and differential pressure from which the inlet flow rate can be derived. The core power generation input includes spatial as well as temporal variations. Some of the more important enhancements provided by VIPRE (as compared to older codes) are an expanded choice of correlations for critical heat flux (CHF) calculations, critical power ratio, two-phase flow and heat transfer for reload and safety analysis, one-pass hot-channel analysis capability, automatic iteration for set point analysis, subcooled voiding capability, and the ability to compute bypass channel flow for boiling water reactor (BWR) applications.

VIPRE-01 predicts the three-dimensional velocity, pressure, thermal energy fields, and fuel rod temperatures for single- and two-phase flow in both pressurized water reactor (PWR) and boiling water reactor (BWR) cores. It solves the finite-difference equations for mass, energy, and momentum conservation for an interconnected array of channels, assuming incompressible thermally expandable homogeneous flow. The equations are solved with no time-step or channel size restrictions for stability. Although the formulation is homogeneous, nonmechanistic models are included for the analysis of subcooled boiling conditions and vapor/liquid slip in two-phase flow. VIPRE-01 is a safety related code and complies with the requirements of 10CFR50 Appendix B.

VIPRE-02 is a thermal-hydraulic analysis code designed to model steady-state conditions and operational transients in LWR cores and vessels for which two phase flow is important. VIPRE-02 uses a two-fluid representation, solving conservation equations for mass, momentum and energy for each phase. This six-equation model is solved implicitly, using a modified Gauss-Seidel iteration procedure, and has no time-step size limitation for stability. Models for phase interaction based on flow regime mapping are provided, using semi-empirical interfacial correlations for heat and mass transfer, and vapor generation.

For core analysis, VIPRE-02 uses a subchannel formulation of the conservation equations; for PWR vessel models, it contains options for a fully three-dimensional representation of the lower plenum. For core analysis, boundary conditions can be specified using the inlet flow or the overall core pressure drop. For vessel models, the hot and cold leg boundaries can be represented using mass sources or pressure sinks in appropriate modes, with local flow blockages to model vessel internal structures and the lower plenum and upper head domes.

The VIPRE code is used for the determination of departure from nucleate boiling (DNB) for those final safety analysis report (FSAR) Chapter 15 transients and accidents for which DNB might be of concern. These events include:

- Steam line break
- Rod withdrawal from subcritical or at power
- Loss of forced reactor coolant flow
- Locked reactor coolant pump rotor or shaft break (PWR) or locked reactor recirculation pump rotor or shaft break (BWR)
- Dropped rod/bank
- Startup of an inactive reactor coolant pump (PWR) or reactor recirculation pump (BWR) for analysis of cold water injection event
- Feedwater malfunction

These events formerly were analyzed using the THINC-IV and FACTRAN codes (Westinghouse). The THINC-IV code performs thermal/hydraulic calculations within the fuel channels, including DNBR evaluation at the clad surface. For calculations in which transient heat conduction within the fuel pins is important, this calculation is performed by FACIRAN. FACTRAN describes the conductive heat transfer within the fuel pin interior and the convective heat transfer at the surface. Iteration may be required between the two codes to obtain convergence on the predicted temperatures. Both the thermal/hydraulic and the conduction/convection calculations are performed simultaneously in VIPRE. In addition to transients listed above, VIPRE can be used for reactor setpoint analysis such as DNBR calculations of core thermal limits for over temperature Delta-T (OT T) trip protection.

Core boundary conditions for VIPRE calculations may be obtained from reactor systems computer codes such as RETRAN or LOFTRAN, a neuronic code such as ANC or SPNOVA and a fuel performance code such as PAD. RETRAN or LOFTRAN provide time-dependent reactor coolant system (RCS) pressure, core average power, core inlet flow rate, and core inlet enthalpy. ANC or SPNOVA provides the core power distribution, nuclear peaking factors and reactivity effects. PAD provides the initial fuel pellet surface temperatures and fuel volume-average temperature.

In Reference [32], VIPRE was found to be particularly sensitive to time-step size in the subcooled boiling region for time-step sizes that exceed the Courant limit based on coolant velocity. Analytical studies have been performed to ensure convergence of the code results when time steps that smaller than the Courant limit are used.

Mixing between the coolant channels is important since turbulence around the hot pin will result in better mixing and in cooler fluid at that location. On the other hand, a lower coolant density in the hot channel might result in crossflow causing coolant displacement to other channels. Coolant mixing between the coolant channels in VIPRE is considered as a combination of turbulent mixing and cross-flow. For example, in the Westinghouse version of VIPRE, a turbulent mixing coefficient is employed. The turbulent mixing coefficient was determined experimentally from a series of subchannel mixing tests conducted by Westinghouse. EPRI found VIPRE results to be relatively insensitive to the value of cross-flow mixing coefficient (Vol. 4 of Ref. [V1]). Additionally, Westinghouse has incorporated the Bishop-Sandberg-Tong heat transfer correlation into VIPRE for post-CHF heat transfer and the Baker-Just correlation for calculation of any zirconium-water reaction that may be calculated to occur.

The Westinghouse VIPRE model requires the following parameters as the input or boundary conditions for calculations:

- Core inlet temperature or enthalpy
- Core exit enthalpy for flow reversal
- Core average power
- Core exit pressure
- Core inlet flow rate
- Core power distributions
- Steady state fuel temperature data

As part of the review process, the NRC staff obtained the VIPRE code from EPRI. Sample inputs for CHF calculation were obtained from Westinghouse. The sample inputs were run on NRC-owned computers. The CHF results were then benchmarked against a separately programmed version of the W-3 and WRB-2 CHF correlations, The CHF results from VIPRE matched those from the separately programmed correlations. In a separate audit calculation, the NRC staff confirmed the conservatism of the void and two-phase friction models selected by Westinghouse by comparison to results calculated using the EPRI default models.

# 4
# PROBABILISTIC RISK ASSESSMENT METHODS AND CODES

The following sections provide descriptions of PRA programs in widespread use for analysis at commercial nuclear power plants. The following codes are included:

- ATHEANA
- CAFTA
- GOTHIC
- EOOS
- HRA CALCULATOR
- SAPHIRE
- PARAGON

## 4.1 ATHEANA

A Technique for Human Event Analysis, (ATHEANA) is a methodology utilized to evaluate and document human failure events (HFEs) for Human Reliability Analysis (HRA) at nuclear power plants. ATHEANA gives insight into how operator responses to equipment failure and accident scenarios impact the probability of core damage (Level 1 PRA) and the release of radioactive material (Level 2 PRA). The ATHEANA methodology is used to do the following:

- Determine possible scenarios where operators are likely to make an error.
- Ascertain possible error forcing contexts (EFCs).
- Develop human error probabilities (HEPs) for incorporation into the probabilistic risk assessment (PRA).
- Analyze possible recovery actions.

Results of the ATHEANA technique are incorporated into Level 1 and Level 2 PRAs and can be used to determine success criteria. They also are used to determine plant procedure changes that may be needed. ATHEANA gives insight as to emergency planning and training needs as well.

The methodology also can be used to identify plant vulnerabilities based on weaknesses in procedures. The methodology addresses digressions from nominal conditions, as well as plant evolutions that may lead to miscommunication or further human performance issues. It also is used to determine performance shaping factors (PSFs) with respect to both nominal and deviation conditions. ATHEANA also analyzes possible interactions among PSFs.

ATHEANA is used to address possible errors of commission (EOC) and errors of omission (EOO) associated with an HEP. The methodology also evaluates further aleatory influences that should be acknowledged for the PRA sequence being evaluated. ATHEANA provides a means for HRA analysts to determine specific operator actions that may occur based on error-forcing contexts. The methodology analyzes the impact of EFCs on an HEP as well as dependencies between multiple HFEs in a sequence. The technique also allows analysts to postulate various plant conditions that could result based on random occurrences of human and equipment failures and successes. ATHEANA also can be used to hypothesize various crew/staff related interactions that could be present at the time of the event that may impact the error rate.

Most HRA methods are evaluation tools used to quantify HEPs. ATHEANA is unique in that it provides HRA guidance and provides an approach to quantify HEPs based on historical data and expert opinion. ATHEANA was developed by the Probabilistic Risk Analysis branch in the Nuclear Regulatory Commission during the 1990s. The methodology was developed in an effort to improve state-of-the-art in the field of HRA. The EPRI HRA Users Group encompasses ATHEANA users as well as other HRA techniques and tools.

### 4.1.1 Capabilities, Computational Structure and Documentation

ATHEANA was developed by the Probabilistic Risk Analysis branch in the Nuclear Regulatory Commission during the 1990s. The methodology was developed in an effort to improve state-of-the-art in the field of HRA. The EPRI HRA Users Group encompasses ATHEANA users as well as other HRA techniques and tools.

- The ATHEANA methodology is used to do the following:
- Determine, model and quantify post-initiator human actions.
- Analyze errors of commission and omission.
- Address possible cognitive and implementation errors for a human action.
- Provide guidance for analyzing a large range of factors relating to the nominal case.
- Provide insight into HFEs that may be risk significant.
- Support effective risk management by identifying enhancements to plant procedures, training and equipment.

The ATHEANA methodology estimates HEPs based on a facilitator-led expert judgment process. This judgment process involves quantifying HEPs on the premise of error-forcing contexts and PSFs that may be set in motion. When estimating the probability of a human failure

event, the facilitator-led expert panel takes into account the likelihood of the EFC occurring in a specific accident scenario, the conditional probability of the unsafe actions that may lead to the HFE and the conditional likelihood that recovery actions will not ensue prior to core damage.

The ATHEANA methodology is documented in the ATHEANA User's Guide [33]. The user's guide provides direction for using ATHEANA to perform human reliability analysis. The user's guide details the qualitative and quantitative analysis approaches encompassed in ATHEANA. The technical basis for ATHEANA [34] discusses the methodology behind ATHEANA and provides guidance for employing the methodology. NUREG-1842 [35] discusses the various HRA methods and how they evaluate when compared to the HRA Good Practices.

### *4.1.2 Range of Applicability, Limitations and Precautions*

As with any other methodology, an understanding of the technique's range of applicability and limitations is essential. The applicability of ATHEANA for human reliability to be incorporated into the PRA is of significant importance. The U.S. Nuclear Regulatory Commission has published the following specific limitations of ATHEANA.

**Table 4-1**
**ATHEANA Limitations**

| Affected Area | Comment |
|---|---|
| HEP Estimates | The HEP estimates are based on expert judgment. As there is no standardized method for evaluating an HEP, the estimates will vary from one expert to another and HEP estimates may be difficult to reproduce. |
| Search Schemes | When developing detailed scenario conditions in order to evaluate relevant influencing factors to be considered in evaluation, search schemes are used. Utilizing these search schemes is resource and time intensive. |
| Developing Performance Shaping Factors (PSFs) | ATHEANA gives a list of various PSFs that can be applied to an HFE. While this gives the analyst flexibility in developing the HFE, it also could lead to variance in results depending on the analyst team if there is no standardized methodology employed. |
| History | While ATHEANA is known in the nuclear industry as a methodology for performing HRA, there are limited documented example applications and expertise surrounding ATHEANA. |
| Pre-Initiator HFEs | While ATHEANA can be used to evaluate pre-initiators, there is no guidance specifically for pre-initiators in ATHEANA. ATHEANA is most practical for analyzing post-initiator HFEs. |

## 4.2 CAFTA

The Computer Aided Fault Tree Analysis System version 5.4, CAFTA5.4, is a computer program used to create, edit and quantify reliability models, utilizing fault trees and event trees. When applied to the nuclear industry, CAFTA5.4 is used to build Level 1 and 2 Probabilistic Risk Assessment (PRA) models. Given a set of initiating events, basic events and operator actions, CAFTA5.4 quantifies the top gate of the fault tree (typically Core Damage Frequency and/or Large Early Release Frequency). CAFTA5.4 is used to do the following:

- Manage, evaluate and print fault trees and event trees
- Generate and analyze cutsets
- Evaluate the influence of events
- Perform risk ranking
- Perform sensitivity analysis

CAFTA5.4 results primarily are used to evaluate the effect of SSCs on Level 1 and 2 PRAs. The results also are used for analyses to support plant modifications, sensitivity analysis, emergency planning and training, and generic plant issue assessments (for example, significance determinations).

CAFTA5.4 is an essential program for the performance of nuclear power plant reliability analysis. Using a set of interactive editors, databases and model quantification tools, analysts are able to quickly modify, evaluate and analyze fault trees. Given a set of user specified initiating events, CAFTA5.4 determines the probability that the top event will occur based on accident sequences specified in the model event trees and the specified success criteria.

CAFTA5.4 interfaces with multiple programs within the R&R Workstation Suite. CQUANT is used to evaluate fault trees in CAFTA and generate cutset results. GTPROB, also part of the R&R Workstation Suite, interfaces with CAFTA and is used to quantify gate probabilities. GTPROB results are updated in the gate (GT) database table and text output files (.GTO) are generated.

### *4.2.1 Capabilities, Computational Structure and Documentation*

CAFTA was developed as part of the Risk & Reliability Workstation, a suite of software tools and applications used for Probabilistic Risk Assessment. The software was developed by the Electric Power Research Institute (EPRI) in 1997. EPRI is charged with maintaining and improving the code. The Risk & Reliability Workstation is supported by the Risk & Reliability (R&R) User's Group. CAFTA has been developed and is maintained under a quality assurance program, which is in compliance with U.S. 10CFR50 Appendix B and ISO 9001 quality assurance requirements.

CAFTA5.4 models events that could occur during an accident. Success criteria and event sequences for initiating events for Level 1 and Level 2 PRAs typically modeled in CAFTA5.4 include the following:

- Anticipated Transient Without Scram (ATWS)
- Interfacing System LOCA (ISLOCA)
- Loss of Coolant Accident (LOCA)
- Loss of Offsite Power (LOOP)
- Loss of Feedwater (LOFW)
- Steam Generator Tube Rupture
- Secondary Line Break Downstream of MSIVs
- Internal Flooding

The CAFTA5.4 database is stored in an ".rr" file. This file contains the basic event (BE), type code (TC) and gate (GT) tables in Microsoft Access format. Data also can be stored in Btrieve format, which makes it compatible with previous versions of CAFTA. CAFTA5.4 can be run on a variety of computer platforms; most commonly PCs. It has the ability to process equations in order to determine event probability, or type code rate. Expressions may be built by typing the equation into the equation field. Event probabilities can either be stored in the reliability database, or equations can be created and stored either in a text file (.txt) or in the .BE file of a CAFTA database. Gate probabilities evaluated using GTPROB gate probabilities are point estimates. CAFTA5.4 has a built in evaluation program to quantify fault trees. However, there are third-party quantification tools that interface with CAFTA, such as FORTE, FTREX, and so on.

The CAFTA5.4 program is documented in the CAFTA User's Manual [36]. The user's manual contains background information regarding CAFTA5.4's capabilities and intended usage. The User's manual also addresses the various configurations of CAFTA for specific applications. A tutorial as well as various filters and utilities CAFTA utilizes is included. Program updates included in CAFTA5.4 also are addressed.

### *4.2.2 Range of Applicability, Limitations and Precautions*

One criterion for producing high quality CAFTA analyses is an understanding of the software's range of applicability and its limitations. Of particular importance is the applicability of the code for generating results that can be used for risk ranking and sensitivity analysis for Level 1 and 2 PRAs. The R&R User's Group has the following program limitations.

**Table 4-2**
**CAFTA5.4 Limitations**

| Affected Area | Comment |
|---|---|
| Evaluation Program | Large reliability models with complex systems can be evaluated with CAFTA5.4, but generally take longer to quantify. Truncating cutsets and purging the database of records is no longer needed and should decrease the quantification time. |
| Fault Tree Editor | Fault trees created and maintained with the educational version of CAFTA are limited to 50 gates within a fault tree. |
| Event Tree Editor | A single event tree is limited to no more than 32,000 sequence end states. |
| Seismic Hazards | Currently, CAFTA does not support dynamic risk models. EPRI is currently working to develop this feature. |

## 4.3 EOOS

The Equipment Out Of Service program Version 3.5, EOOS3.5, is a computer program used for monitoring safety at industrial facilities. EOOS3.5 is a Configuration Risk Management (CRM) tool that uses a safety or risk model of the plant, based on fault tress and minimal cutsets, such as those developed in a Probabilistic Risk Assessment (PRA). When applied to the nuclear industry, EOOS3.5 monitors the availability of important structures, systems and components (SSCs).

Based on the current plant configuration, EOOS3.5 can propagate information through the PRA model and quantify risk measures specified in the model. Fault tree results are translated into color-coded status panels, lists of risk-significant and applicable activities. Users of EOOS3.5 can easily identify safety problems, and specific work activities that cause it. This risk assessment analysis allows the user to determine if the scheduled work activity warrants special contingency actions based on increased risk.

EOOS3.5 is used to do the following:

- Propagate information through the PRA model based on plant configuration.
- Quantify risk measures specified in the model.
- Translate fault tree results into lists of risk-significant and relative activities, as well as color-coded status panels.
- Allow the user to easily determine if the scheduled work activity warrants special contingency actions based on increased risk.

EOOS 3.5 results primarily are used to determine which structures, systems and components should remain available while performing scheduled work activities based on risk metrics. The program allows plants to determine what improvements should be made with respect to plant

safety based on risk significance. The software allows users to update work schedules as plant configuration changes occur (SSCs become unavailable). Work schedules can be adjusted based on a (4) results. This allows users to avoid operational mistakes that could be costly.

EOOS3.5 minimizes conservative planning requirements thereby reducing operation and maintenance costs. EOOS3.5 results allow users to minimize outage durations.

### *4.3.1 Capabilities, Computational Structure and Documentation*

EOOS was developed as part of the Risk & Reliability Workstation, a suite of software tools and applications used for Probabilistic Safety Assessment (PSA). The software was developed by the Electric Power Research Institute (EPRI) in 1992. EPRI is charged with maintaining and improving the code. More than 70 plants worldwide currently employ EOOS. The Risk & Reliability Workstation is supported by the Risk & Reliability (R&R) User's Group. EOOS has been developed and is maintained under a quality assurance program, which is in compliance with U.S. 10CFR50 Appendix B quality assurance requirements.

EOOS3.5 users are able to:

- Quickly determine risk level based on current plant configuration.
- Plan future work schedules so that risk is minimized.
- Analyze which SSCs are important and driving the risk level.
- Reduce maintenance and operating costs by minimizing unavailability.
- Interface with other programs (for example, CAFTA, Forte, FTRex, and so on).
- Evaluate external events via the PRA model and a qualitative approach.

The relationship data in previous versions of EOOS was contained in Dbase (.DBF) files. The relationship data in EOOS 3.5 is contained in a Microsoft Access database. EOOS3.5 can be run on a variety of computer platforms, most commonly PCs. Most of the basic event probability equations within the program are point estimates. However, there are some event probabilities which vary with plant conditions. In these instances event probabilities are expressed using formulas. Equations for event probability are stored either in a text file (.txt) or in the .BE file of a CAFTA database.

The EOOS3.5 code is documented in the EOOS User's Manual [37]. The user's manual contains detailed descriptions of the capabilities and functions of EOOS3.5. The user's manual also gives direction to operators and schedulers at nuclear power plants as to how they can use EOOS3.5 to meet their risk management needs.

### 4.3.2 Range of Applicability, Limitations and Precautions

One criterion for producing high quality EOOS3.5 analyses is an understanding of the software's range of applicability and its limitations. Of particular importance is the applicability of the program to determine high risk configurations. The following program limitations are noted within the EOOS User's Manual.

**Table 4-3**
**EOOS3.5 Limitations**

| Affected Area | Comment |
|---|---|
| Quantification | Conservatisms may be produced in results as EOOS does not subsume cutsets used to determine risk importance values for SSCs (both out-of-service and in-service). |
| Quantification | EOOS uses .BAT files for cutset quantifiers as opposed to command-line programs. This difference raises a few limitations on the setup of EOOS. The command line length is limited and can be overstepped if EOOS data is saved in folders with lengthy path names. To navigate around this limitation users should ensure that folders where EOOS software and data are stored are as short as possible. |
| Quantification | EOOS can be configured to evaluate multiple top events; however this configuration may not work with some fault tree evaluation programs. |
| Mutually Exclusive Events | Mutually exclusive events specified in the PSA list may be inappropriate for EOOS. |

## 4.4 HRA Calculator

The Human Reliability Analysis Calculator Version 4.0 (HRAC4.0) is a computer program used for human reliability analysis (HRA). HRAC4.0 is used to create and record human failure events (HFEs). The program also is utilized to quantify human error probabilities (HEPs). HRAC4.0 is used to do the following:

- Provide qualitative information regarding operator response to accident sequences.
  - HEPs can be included in PRA fault tree and event tree models to realistically measure the impact of operator actions in mitigating accident sequences.
- Provide quantitative estimates regarding the probability that a human error will occur when attempting to mitigate accidents, as well as when operators are performing maintenance and testing of structures, systems and components (SSCs).
- Evaluate the effect of operator actions.

- Predict if operator and plant personnel have sufficient time to complete the actions necessary to prevent an accident from occurring.
- Provide insight as to where operator training and plant procedure changes may be needed in order to successfully mitigate an accident.

HRAC4.0 results are used primarily to evaluate accident sequence models. The results also are used to determine what operator actions may contribute to a risk significant event. HRAC4.0 results allow plants to determine if changes in plant procedures need to be made.

HRAC4.0 is an integral part of developing an accurate PRA model. It allows PRA analysts to fully examine and understand the effect operator interactions with plant systems have on accident sequences. As human errors may lead to a degradation of performance as well as failure, HRA is an important aspect of PRA modeling.

This new version of the HRAC4.0 allows analysts to implement a combination method for evaluating the probability of cognition (Pcog) using combinations of other methods (see list of methodologies subsection below for acronyms) such as Max of CBDTM and HRC/ORE, CBDTM +ASEP or CBDTM+HCR/ORE, as well as define CBDTM probabilities and apply multiple recoveries for CBDTM. HRAC4.0 interfaces with PRA DocAssist, has a spell checker and provides additional documentation fields for better alignment to the ASME PRA Standard.

### *4.4.1 Capabilities, Computational Structure and Documentation*

The EPRI HRA/PRA Tools User Group is charged with the development of the HRA Calculator. The EPRI HRA/PRA Tools User Group was established in 2004 and currently encompasses 19 utilities embodying more than 60 US nuclear power plants, as well as one international member. The goal of the HRA/PRA Tools User Group is to aid the nuclear industry in approaching standardized HRA methods.

HRAC4.0 employs the following methodologies when evaluating HEPs:

- Human Cognitive Reliability/Operator Reactor Experiments (HCR/ORE)
- Techniques for Human Error Rate Prediction (THERP)
- EPRI's Cause-Based Decision Tree Method (CBDTM)
- Accident Sequence Evaluation Program Human Reliability Analysis Procedure (ASEP)
- Standardized Plant Analysis Risk - Human (SPAR-H)

The HRA Calculator is a stand-alone program and can be run on a variety of PC platforms. The HRAC interfaces with two Microsoft Access databases. One database is created by the user and stores analysis data. The other database stores invariable program data (for example, CBDTM trees, THERP tables, and so on), which contains underlying logic used to compute human error probability.

The format of the HRAC is tailored to reliability analysts. The equations in the program are essentially lumped parameter equations. The HRAC4.0 can quickly evaluate Human Error Probabilities (HEPs).

The HRAC4.0 program is documented in "The EPRI HRA Calculator® Software Users Manual, Version 4.0" [38]. The user's guide contains detailed descriptions of the required input and methodologies utilized to evaluate HEPs.

### 4.4.2 Range of Applicability, Limitations and Precautions

An understanding of the HRA Calculator's range of applicability and its limitations is necessary in order to produce quality results. Of particular importance is the applicability of the program to improve the integrity of PRA models as well as identify deficiencies in plant procedures or performance shaping factors. The following limitations have been noted regarding HRAC4.0.

**Table 4-4**
**HRA Calculator 4.0 Limitations**

| Type | Comment |
|---|---|
| Program bug | Program frequently quits when generating report, using shortcut commands or when right clicking |
| Report generation | HEP report still needs to be manipulated when generated, especially when using combined assessment methods. HEPs with screening values do not allow users to enter text for documentation purposes. |
| Higher execution error probabilities | Stress multipliers and no recovery credit for procedure checks result in higher execution error probabilities. |
| Inconsistent mean values | The HRAC uses mean values when THERP is used, but not when ASEP or CBDTM is used. |
| Performance Shaping Factors (PSF) | Not all Performance Shaping Factors are handled within HRAC. Since the PSFs are standardized within the HRAC, there is not much flexibility for inputting factors that are not in the program. |

## 4.5 SAPHIRE

The System Analysis Programs for Hands-on Integrated Reliability suite Version 7.0 (SAPHIRE7.0) is a suite of computer programs designed with an objective of allowing analysts at nuclear power plants to develop and evaluate probabilistic risk assessment (PRA) models. The programs included in the SAPHIRE suite are the System Analysis and Risk Assessment (SARA) system, the Fault tree, Event tree and P&ID (FEP) editors, the Integrated Reliability and Risk Analysis System (IRRAS) and the Models and Results Database (MAR-D). SAPHIRE7.0 is used to do the following:

- Model nuclear power plant's response to initiating events.
- Evaluate external and internal initiating events.

- Determine significant contributors to core damage (Level 1 PRA).
- Evaluate containment behavior during severe accident scenarios which can lead to radioactive releases (Level 2 PRA).
- Evaluate risk with regard to release consequences to the public (Level 3 PRA).
- Perform uncertainty analysis utilizing the Monte Carlo and Latin Hypercube methods.
- Perform probabilistic risk assessment during full power, low power or shutdown conditions at nuclear power plants.
- Convert an internal events model into an external events model.

SAPHIRE results are utilized to determine Level 1, 2 and 3 success criteria for probabilistic risk assessments (PRAs). The results also are used for equipment qualification analyses, sensitivity analyses, core damage frequency (CDF) and large early release frequency (LERF) determinations, generic plant issue assessments (for example, significance determinations) and other similar applications.

The U.S. Nuclear Regulatory Commission (NRC) utilizes SAPHIRE for activities associated with PRA including advanced reactor certification, resolving generic issues, low power and shutdown studies, inspections, developing risk-informed regulation guidance and quantifying operator actions.

Previous versions of SAPHIRE consisted of only the FEP, SARA and MAR-D modules. Utilizing the Windows 95 (or Windows NT) environment, all of the modules were incorporated in SAPHIRE versions 6 and 7. These versions also were updated to include the Graphical Evaluation Module (GEM). GEM automates the process of quantifying operator actions at nuclear power plants. GEM implements Accident Sequence Precursor (ASP) methods.

Many aspects of SAPHIRE have been enhanced in version 7.0. The graphical and fault tree logic editors have been improved to simplify creating and analyzing data. Using the Advanced Rules Editors, full scale programming languages may be employed to create partition, linkage and recovery rules. The development of a new interface, the Sensitivity Wizard, allows users to perform importance analysis. SAPHIRE7.0 is capable of evaluating projects with up to 100,000 sequences.

### *4.5.1 Capabilities, Computational Structure and Documentation*

SAPHIRE is funded by the U.S. Nuclear Regulatory Commission (NRC) and developed by Idaho National Laboratory (INL). The INL released the initial version of SAPHIRE, the "Integrated Risk and Reliability Analysis System" (IRRAS), in February 1987.

State-of-the-art constraints drove the INL to develop independent modules that complimented the capabilities of IRRAS. These modules are Fault Tree, Event Tree, and Piping and Instrumentation Diagram (FEP), System Analysis and Risk Assessment (SARA) and Models and Results Database (MAR-D). The INL currently is charged with developing and maintaining SAPHIRE. SAPHIRE has been distributed to thousands of users including NRC staff,

engineering firms, national laboratories, utilities, industry contractors, architectural engineering firms and other government agencies.

The SAPHIRE User's Group provides technical support and training for SAPHIRE as well as regular program updates. SAPHIRE is developed and maintained under a quality assurance program. A verification and validation test of SAPHIRE5.0 was performed in 1987 and is documented in NUREG/CR-6116, Volume 9, Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 5.0, Verification and Validation (V&V) Manual, 1995 [39].

SAPHIRE's relational database allows users to seamlessly create fault and event trees, evaluate accident scenario event trees and system fault trees, perform uncertainty and sensitivity analysis, define basic event failure and accident sequence data, evaluate cutsets, record results and develop reports. Utilizing SAPHIRE for PRA analysts are able to:

- Create and modify event trees using a graphical editor.
- Quantify large event trees using the large event tree methodology.
- Generate cutsets for large fault trees and accident sequences.
- Analyze cutsets using the slice options.
  - The slice by event option allows analysts to dissect cutsets into two lists based on user-specified quantification.
  - The slice by rule option allows analysts to dissect cutsets into two lists based on user-specified quantification.
- Assign dynamic flag sets to a sequence(s) via event tree rules based on criteria specified in the rule.
- Remove mutually exclusive events by editing cutsets manually, utilize the mutually exclusive top event attribute within SAPHIRE, apply recovery rules, prune cutsets or edit logic models.
- Quantify operational events using NRC's Standardized Plant Analysis Risk (SPAR) models and methods via GEM.TBD.

The SAPHIRE7.0 program is written in Modula-2 and can be run on a variety of computer platforms; most commonly PCs. SAPHIRE7.0 employs a relational database to control inputs and outputs. The program consists of nine relations, or tables, where PRA data is stored and retrieved. The SAPHIRE7.0 database is separated into projects.

The method used to evaluate fault trees is essentially top down algorithms. Point estimates initially are calculated for top events in fault trees. Users have the option of choosing from three top event quantification methods within SAPHIRE: the min-max method, minimal cutset upper bound method or the rare event approximation. The program uses standard methods for quantification and propagation of failure information through cutsets generated. SAPHIRE7.0 interfaces with other PRA codes via the ASCII-based input/output capability of the Models and Results Database (MAR-D) module.

The SAPHIRE7.0 suite is documented in the SAPHIRE7.0 Reference Manual [40]. The reference manual contains a user's guide as well as a technical reference section. The user's guide provides an overview of SAPHIRE's capabilities, special features of the program and how SAPHIRE can be used for PRA. The technical reference section details the probabilistic and mathematical concepts used in the software. Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 5.0, Verification and Validation (V& V) Manual [SA1] details the quality assurance program performed with respect to SAPHIRE software development.

### *4.5.2 Range of Applicability, Limitations and Precautions*

One criterion for producing high quality SAPHIRE7.0 analyses is an understanding of the code's range of applicability and its limitations. Of particular importance is the applicability of the code for generating results that can be used to determine success criteria. The SAPHIRE Reference manual lists the following specific limitations within the program.

**Table 4-5**
**SAPHIRE7.0 Limitations**

| Affected Area | Comment |
|---|---|
| Level 3 PRA | SAPHIRE7.0 has limited capabilities for Level 3 PRA analysis |
| Database Limits | The following are basic database limits noted within SAPHIRE:<br>• Number of basic events 64,000<br>• Number of event trees 64,000<br>• Number of sequences 2,000,000<br>• Number of end states 64,000<br>• Number of fault trees 64,000 - # of basic events<br>• Number of gates 64,000<br>• Number of change sets 10,000<br>• Number of analysis types 16<br>• Inputs per OR gate 5,000<br>• Inputs per AND gate 256<br>• Inputs per N/M gate N/ 99, where N = 2 to 98.<br>• Gates/events per fault tree 10,000 (Limit when loading a fault tree through MAR-D).<br>• Number of events per cut set 256<br>• Number of uncertainty samples 99,999<br>• Alpha to Graphics conversion 50 levels deep<br>• Logic display 255 characters wide (dialog display and reporting only)<br>• Path search display 255 characters wide (dialog display and reporting only) |

## 4.6 PARAGON

The PARAGON® software version 1.3, PARAGON1.3, is a multi-purpose tool intended for use by nuclear plant PRA and non-PRA personnel. PARAGON provides users a comprehensive ability to implement Configuration Risk Management (CRM) programs to comply with 10CFR 50.65(a)(4), for all modes of operation, using a blended approach to risk assessments.

PARAGON is a software tool for risk-informed decision-making. PARAGON is designed to expedite the assimilation of plant management activities and assist with long-term implementation of efficiency measures within the facility. The PARAGON software package provides a complete framework for risk-informed decision-making for configuration risk management and other risk-informed decisions.

The software is used to do the following:

- Provide qualitative measures of safety using Decision Trees (for example, Safety Functions, Transients, and so on).
- Provide quantitative measure of risk via:
  - PRA Engine interface (for example, CAFTA, WinNupra, RiskSpectrum).
  - Probabilistic Shutdown Safety Assessment (PSSA).
- Provide the ability to combine these quantitative results with deterministic and other qualitative information.
- Allow for plant-specific thresholds and provide the ability to assign safety classifications (via colors) and values based on absolute or relative risk increase.
- Allow for real-time evaluation of risk status.
- Evaluate external events via the PRA model (using standard event tree/fault tree modeling tools such as CAFTA, WinNupra, RiskSpectrum) and assessment trees.

The setup of a PARAGON model to perform CRM typically includes a combination of qualitative and quantitative features with customized results displayed for PRA personnel, operators and work planning users.

For Online operation the internal events PRA (for example, CDF/LERF, Cumulative, Remain in Service/Return to Service (RIS/RTS) Lists and graphic profiles are used for quantitative measures of risk; while qualitative measures of risk are provided via guidance and defense-in-depth (safety functions, plant transients, support systems, RIS/RTS Lists). For Outage operation the shutdown PRA/PSSA (for example, CDF/RCS Boiling Frequency/LERF, Cumulative, RIS/RTS Lists), thermal hydraulic calculations (for example, decay heat level, time-to-boil, time-to-core damage) and graphic profiles are used to provide quantitative measures of risk; while qualitative measures of risk are provided via guidance and defense-in-depth (safety functions, support systems, RIS/RTS Lists).

### 4.6.1 Capabilities, Computational Structure and Documentation

ERIN Engineering and Research, Inc. (ERIN), an SKF Group Company, developed the PARAGON software. The PARAGON program builds upon the EPRI ORAM-SENTINEL technology. ERIN is responsible for maintaining and improving the software, which has been developed and maintained under an Appendix B quality assurance program.

Using a combination of features, PARAGON1.3 provides means to assimilate external risk into a CRM Model. PARAGON1.3 has the following capabilities:

- Provide qualitative configuration risk assessments.
- Evaluate increased probability of initiators.
- Perform integration of PRA model and PRA engine.
- Provide text-based guidance.
- Support user-defined limits and displays.

PARAGON is a Network-based program running on an Enterprise Database (J2EE Application Server) that runs with a database server (Oracle or SQL Server). PARAGON is written in JAVA and can be run on a variety of computer platforms. The PARAGON model is made plant-specific. Java Native Interface (JNI) is used to communicate with non-Java external applications. PARAGON is stand-alone capable using personal/express editions of the database.

The PARAGON Help File [41] can be used to navigate the software and details the functions and features of PARAGON. The PARAGON1.3 Getting Started Manual [42] directs the user through installation of the PARAGON software.

The PARAGON1.3 Getting Started manual [42] directs the user through the setup of the PARAGON data structures, the installation of PARAGON programs (Server, Client, Conversion), and through the performance of an Installation Acceptance Test. The manual includes instructions to install the PARAGON components on a new machine and also for upgrading from previous versions of PARAGON. Upon the successful completion of the actions outlined in the Getting Started manual, the PARAGON environment is established for its use.

### 4.6.2 Range of Applicability, Limitations and Precautions

An understanding of PARAGON's range of applicability allows analysts to comply with 10CFR 50.65(a)(4), by managing risk for all modes of operation. Of particular importance is the applicability of the program to provide defense-in-depth evaluations by modeling safety functions and plant transients based on success criteria.

An electronic knowledge base of program issues is accessible by PARAGON users via the internet. This knowledge base addresses program bugs, workarounds and technical issues.

PARAGON models are built by the user. If the PARAGON model is inaccurate, accurate estimates of plant risk levels cannot be obtained.

# 5
# CONCLUSIONS AND RECOMMENDATIONS

Due to multiple reasons (reduced greenhouse gas emissions, energy security, economic competitiveness, and so on), it is imperative that the current fleet of commercial nuclear power plants continues to operate at high performance levels into at least the middle decades of the 21st century. Although the current level of performance of the fleet is excellent, it is not certain that it can be sustained over the long future periods of operation that are envisioned. At the time of the original licensing of these plants, safety margins that were adequate to protect the health and safety of the public were established. This was achieved by performing conservative engineering analyses and using conservative judgment in specifying appropriate safety limits for critical plant parameters. However, over the long period of envisioned service, operation of the plants has the potential to impact these safety margins. Operational changes made to enhance plant economics, such as power uprates, also can have an impact on these margins. Thus, there is an explicit need for plants to assess and manage safety margins over this period of extended operation. While technically sound, the current analytical approaches and tools that are used to assess safety margins are inefficient and labor intensive to apply. Additionally, technological advances being pursued to enhance long term plant operation may require analyses that are outside of the bounds under which some of the current safety analysis models have been validated.

The research described in this report evaluated the current state of the art with respect to evaluation of nuclear power plant safety margins. It also reviewed methods and tools that are in widespread use to perform plant safety analyses and probabilistic risk assessments for their applicability to address safety margin issues. Currently, no consensus approach exists that is both accurate and efficient to evaluate and manage nuclear plant safety margins. For performance of safety margin evaluations the current generation of safety analysis and probabilistic risk assessment tools also are not sufficiently integrated to permit efficient analysis. Additionally, enhancements to some safety analysis tools will be necessary to permit evaluation of postulated plant enhancements that could be implemented to support long term plant operation.

In the development of an approach to assess the impact of operational and design changes on safety margins, the Nuclear Energy Agency Committee on the Safety of Nuclear Installations Safety Margins Working Group developed and tested a proposed approach to address the following issues:

- Develop a working definition of safety margins and related concepts,
- Develop a process for the assessment of safety margins,
- Identify appropriate methods for safety margin evaluation,

- Identify methods for safety margin quantification,
- Prepare a guidance document on safety margins for use by regulatory authorities.

This proposed approach was validated on two hypothetical case studies. A review of these case studies indicates that, although the proposed process can provide useful and informative results for the evaluation of the impact of plant changes on safety margins, it appears that it will be very complex and labor intensive to perform on actual applications. Thus, it can be concluded that significant additional work will be required to develop a method that provides technically acceptable results and is practical to implement.

The evaluation of the current suite of DSA and PRA tools indicated that there are significant limitations and inefficiencies in the integration of the DSA and PRA technologies as they currently exist and are applied. In particular, application of the current state of the art can be characterized as a "brute force" approach that requires a sequence of analyses to evaluate and characterize safety margins. Furthermore, the tools and methods have legacy issues from their initial development 20 to 40 years ago; that is, they have simplified mechanistic models, inefficient computational algorithms, primitive connectivity, and non-intuitive visualization of results. Although adequate to meet current regulatory requirements and operational needs, application of the current technology is usually overly conservative, labor intensive, time consuming and expensive. As a result, potential applications for simulation and monitoring often are precluded.

As a result of these assessments, it can be concluded that the objective of performing effective and efficient risk informed safety margin characterizations will require developing an integrated suite of methods and tools to seamlessly perform the individual tasks associated with these evaluations. This suite of tools will need to provide the capability to perform both physics based analyses (for example, integrated neutron kinetics/computational fluid dynamics/thermal-hydraulics/and so on) and plant risk assessments. To achieve these objectives, it is recommended that the following research activities be performed.

1. Develop an integrated and practical framework for performing risk informed safety margin characterization. This task will utilize insights obtained during this research effort. It also is recommended to integrate this research with research that is planned to be conducted by the U. S. Department of Energy (DoE) funded Light Water Reactor Sustainability (LWRS) program [43].
2. Based on the research described in this report, it clearly would be very beneficial (and for some possible plant enhancements, even necessary) to develop the next generation of integrated safety analysis tools. As part of the LWRS research program, the Idaho National Laboratory is embarking on a multi-year development effort to develop such an integrated safety analysis tool. This development effort (known as R7) was initiated in 2009 and will, over the next several years, develop an integrated suite of safety analysis tools. The R7 suite is intended to leverage advances in computer hardware, programming technology, numerical methods, data management and visualization technology to support accurate and efficient analysis by a broad spectrum of users.

3. Similarly, for risk assessment technology, it also would be beneficial to develop an integrated suite of tools to perform efficient and effective risk assessments and risk management activities. In response to this need, EPRI initiated investigation into such a tool (Phoenix). It is recommended that development of this tool be performed and include the capability to interface with applicable DSA tools (including R7). Note that a functional specification for the Phoenix PRA software has been developed and is available in EPRI report 1019207 [44].

Thus, the planned research and development can be viewed as consisting of three inter-related groups addressing the following issues: (1) advanced DSA techniques, (2) advanced PRA techniques, and (3) methods/tools for analysis, integration and visualization of results to support effective and efficient decision-making. For all three groups, the research is intended to address completeness of analysis, treatment of uncertainty, and efficiency of computation so that more accurate and cost-effective techniques can be used to address safety margin characterizations.

# 6
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# A
# APPENDIX: KEY NUCLEAR POWER PLANT SAFETY LIMIT PARAMETERS

This appendix provides a listing of plant parameters identified to be critical to NPP safety. These parameters we identified from existing regulatory requirements in the United States (for example, 10CFR) and NPP Standard Technical Specifications (as specified in NUREGs 1430 – 1434).

## Nuclear Fuel

### *ECCS Acceptance Criteria*

1. *Peak Cladding Temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200º F. (Ref. 10CFR50.46 (b)(1))
2. *Maximum Cladding Oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. (Ref. 10CFR50.46 (b)(2))
3. *Maximum Hydrogen Generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. (Ref. 10CFR50.46 (b)(3))
4. *Coolable Geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling. (Ref. 10CFR50.46 (b)(4))
5. *Long-Term Cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. (Ref. 10CFR50.46 (b)(5))

### *BWR Fuel Integrity Safety Limits*

1. *Thermal Power* shall be ≤ 25% rated thermal power (RTP) whenever reactor steam dome pressure < 785 psig or core flow < 10% rated core flow. (Ref. NUREG 1433 BWR4 STS/ NUREG 1434 BWR6 STS – Section 2.1.1.1 (Safety Limits))
2. *Minimum Critical Power Ratio* (MCPR) shall be ≥1.07 for two recirculation loop operation or ≥1.08 for single recirculation loop operation whenever the reactor steam dome pressure ≥785 psig and core flow ≥10% rated core flow. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 2.1.1.2 (Safety Limits))

3. *Reactor Vessel Water Level* shall be greater than the top of active irradiated fuel (TAF). (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 2.1.1.3 (Safety Limits))
4. *Shutdown Margin* shall be (a) ≥0.38% Δk/k with the highest worth control rod analytically determined or (b) ≥0.28% Δk/k with the highest worth control rod determined by test. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 3.1.1 (Reactivity Control Limits))
5. All *Average Planar Linear Heat Generation Rates* (APLHGR) shall be less than or equal to the limits specified in the COLR. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 3.2.1 (Reactivity Control Limits))
6. All *Minimum Critical Power Ratios* (MCPR) shall be less than or equal to the limits specified in the COLR. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 3.2.2 (Reactivity Control Limits))
7. All *Linear Heat Generation Rates* (LHGR) shall be less than or equal to the limits specified in the COLR. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 3.2.3 (Reactivity Control Limits))
8. *Maximum Fraction of Limiting Power Density* (MFLPD) shall be less than or equal to Fraction of RTP. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 3.2.4 (Reactivity Control Limits))

### *PWR Fuel Integrity Safety Limits – Babcock and Wilcox Plants*

1. *Maximum Local Fuel Pin Centerline Temperature* shall be ≤5080 – (6.5 x 10-3 MWD/MTU) °F in Modes 1 and 2 (B&W plants). (Ref. NUREG 1430 B&W STS – Section 2.1.1.1 (Safety Limits))
2. *Departure from Nucleate Boiling Ratio* (DNBR) shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation in Modes 1 and 2. (Ref. UREG 1430 B&W STS – Section 2.1.1.2 (Safety Limits))
3. *Reactor Coolant System* (RCS) *Core Outlet Temperature and Pressure in* Modes 1 and 2, shall be maintained above and to the left of the SL shown in Figure 2.1.1-1 of STS. (Ref. NUREG 1430 B&W STS – Section 2.1.1.3 (Safety Limits))
4. *Shutdown Margin* shall be within the limits specified in the COLR (Modes 3 – 5). (Ref. NUREG 1430 B&W STS – Section 3.1.1 (Reactivity Control Limits))
5. *Moderator Temperature Coefficient* (MTC) shall be maintained within the limits specified in the COLR. (Ref. NUREG 1430 B&W STS – Section 3.1.3 (Reactivity Control Limits))
6. *Axial Power Imbalance* shall be maintained within the limits specified in the COLR. (Ref. NUREG 1430 B&W STS – Section 3.2.3 (Reactivity Control Limits))
7. *Quadrant Power Tilt* (QPT) shall be maintained less than or equal to the steady state limits specified in the COLR. (Ref. NUREG 1430 B&W STS – Section 3.2.4 (Reactivity Control Limits))
8. *Power Peaking Factors* shall be within the limits specified in the COLR. (Ref. NUREG 1430 B&W STS – Section 3.2.5 (Reactivity Control Limits))

### *PWR Fuel Integrity Safety Limits – Combustion Engineering Plants*

1. *Combination of Thermal Power, Pressurizer Pressure, and the Highest Operating Loop Cold Leg Coolant Temperature* in Modes 1 and 2, shall not exceed the limits shown in Figure 2.1.1-1 of STS. (Ref. NUREG 1432 CE STS – Section 2.1.1.1 (Safety Limits))
2. *Peak Fuel Centerline Temperature* shall be maintained at < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-275-P, Revision 1-P-A or CENPD-382-P-A in Modes 1 and 2. (Ref. NUREG 1432 CE STS – Section 2.1.1.2 (Safety Limits))
3. *Shutdown Margin* shall be within the limits specified in the COLR (Modes 3 – 5). (Ref. NUREG 1432 CE STS – Section 3.1.1 (Reactivity Control Limits))

### *PWR Fuel Integrity Safety Limits – Westinghouse Plants*

1. *Combination of Thermal Power, Reactor Coolant System (RCS) Highest Loop Average Temperature, and Pressurizer Pressure* in Modes 1 and 2 shall not exceed the limits specified in the COLR. (Ref. NUREG 1431 W STS – Section 2.1.1 (Safety Limits))
2. *Departure from Nucleate Boiling Ratio* (DNBR) shall be maintained ≥1.17 for the WRB-1/WRB-2 DNB correlations. (Ref. NUREG 1431 W STS – Section 2.1.1.1 (Safety Limits))
3. *Peak Fuel Centerline Temperature* shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup. TBD
4. *Shutdown Margin* shall be within the limits specified in the COLR (Mode 2 with $k_{eff}$ < 1.0, and Modes 3-5). (Ref. NUREG 1431 W STS – Section 3.1.1 (Reactivity Control Limits))

## Reactor Pressure Vessel and Primary System

### *BWR Plants*

1. *Reactor steam dome pressure* shall be ≤1325 psig. (Ref. NUREG 1433 BWR4 STS/NUREG 1434 BWR6 STS – Section 2.1.2 (Safety Limits))
2. *RCS Pressure, Temperature, Heatup/Cooldown Rates, and Recirculation Pump Starting Temperature* requirements shall be maintained within the limits specified in the PTLR. (Ref. NUREG 1433 BWR4 STS – Section 3.4.10/NUREG 1434 BWR6 STS – Section 3.4.11 (Reactor Coolant System))
3. *Reactor Steam Dome Pressure* shall be ≤1020 psig (BWR4)/1045 psig (BWR6). (Ref. NUREG 1433 BWR4 STS – Section 3.4.11/NUREG 1434 BWR6 STS – Section 3.4.12 (Reactor Coolant System))

### PWR – Babcock and Wilcox Plants

1. *Reactor coolant system* (RCS) *pressure* shall be ≤2750 psia in Modes 1–5. (Ref. NUREG 1430 B&W STS – Section 2.1.2 (Safety Limits))
2. Maintain *RCS DNB Parameters* for *Loop Pressure*, *Hot Leg Temperature*, and *RCS Total Flow Rate* within the following limits in Mode 1: (a) with four reactor coolant pumps (RCPs) operating RCS loop pressure shall be ≥2061.6 psig, RCS hot leg temperature shall be ≤604.6°F, and RCS total flow rate shall be ≥139.7 E6 lb/hr, and (b) with three RCPs operating RCS loop pressure shall be ≥2057.2 psig, RCS hot leg temperature shall be ≤604.6°F, and RCS total flow rate shall be ≥104.4 E6 lb/hr. (Ref. NUREG 1430 B&W STS – Section 3.4.1 (Reactor Coolant System))
3. Each *RCS Loop Average Temperature* ($T_{avg}$) shall be ≥525°F in Mode 1 or Mode 2 with $k_{eff}$ ≥1.0. (Ref. NUREG 1430 B&W STS – Section 3.4.2 (Reactor Coolant System))
4. *RCS Pressure, Temperature, and Heatup/Cooldown Rates* shall be maintained within the limits specified in the PTLR. (Ref. NUREG 1430 B&W STS – Section 3.4.3 (Reactor Coolant System))
5. The pressurizer shall be operable with (a) *Pressurizer Water Level* ≤290 inches and (b) minimum of 126 kW of *Pressurizer Heater Power* operable and capable of being powered from an emergency power supply. (Ref. NUREG 1430 B&W STS – Section 3.4.9 (Reactor Coolant System))

### PWR – Combustion Engineering Plants

1. *Reactor coolant system* (RCS) *pressure* shall be ≤2750 psia in Modes 1–5. (Ref. NUREG 1432 CE STS – Section 2.1.2 (Safety Limits))
2. Maintain RCS DNB Parameters for Pressurizer Pressure, Cold Leg Temperature, and RCS Total Flow Rate within the following limits in Mode 1: (a) *Pressurizer Pressure* ≥2025 psia and ≤2275 psia, (b) *RCS Cold Leg Temperature* ($T_c$) ≥535°F and ≤558°F for < 70% RTP or ≥544°F and ≤588°F for ≥ [70]% RTP, and (c) *RCS Total Flow Rate* ≥ 148 E6 lb/hour. (Ref. NUREG 1432 CE STS – Section 3.4.1 (Reactor Coolant System))
3. Each RCS Loop Average Temperature ($T_{avg}$) shall be ≥520°F in Mode 1 with $T_{avg}$ in one or more RCS loops < 535°F and Mode 2 with $T_{avg}$ in one or more RCS loops < 535°F and $k_{eff}$ ≥1.0. (Ref. NUREG 1432 CE STS – Section 3.4.2 (Reactor Coolant System))
4. *RCS Pressure, Temperature, and Heatup/Cooldown Rates* shall be maintained within the limits specified in the PTLR. (Ref. NUREG 1432 CE STS – Section 3.4.3 (Reactor Coolant System))
5. The pressurizer shall be operable with (a) P*ressurizer Water Level* < 60% and (b) pressurizer heaters operable with *Pressurizer Heaters Capacity* ≥150 kW and capable of being powered from an emergency power supply. (Ref. NUREG 1432 CE STS – Section 3.4.9 (Reactor Coolant System))

### *PWR – Westinghouse Plants*

1. *Reactor coolant system* (RCS) *pressure* shall be ≤2735 psig in Modes 1–5 (W plants). (Ref. NUREG 1431 W STS – Section 2.1.2 (Safety Limits))
2. Maintian RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature, and RCS Total Flow Rate within the following limits (a) *Pressurizer Pressure* is greater than or equal to the limit specified in the COLR, b. *RCS Average Temperature* is less than or equal to the limit specified in the COLR, and c. *RCS Total Flow Rate* ≥284,000 gpm and greater than or equal to the limit specified in the COLR. (Ref. NUREG 1431 W STS – Section 3.4.1 (Reactor Coolant System))
3. Each *RCS Loop Average Temperature* ($T_{avg}$) shall be ≥541°F in Mode 1 or Mode 2 with $k_{eff}$ ≥1.0. (Ref. NUREG 1431 W STS – Section 3.4.2 (Reactor Coolant System))
4. *RCS Pressure, Temperature, and Heatup/Cooldown Rates* shall be maintained within the limits specified in the PTLR. (Ref. NUREG 1431 W STS – Section 3.4.3 (Reactor Coolant System))
5. The pressurizer shall be operable with (a) *Pressurizer Water Level* • 92% and (b) pressurizer heaters operable with *Pressurizer Heaters Capacity* ≥125 kW. (Ref. NUREG 1431 W STS – Section 3.4.9 (Reactor Coolant System))

## Containment

### *BWR-4 Plants*

1. *Drywell Pressure* shall be ≤0.75 psig (Applicable to Modes 1–3). (Ref. NUREG 1433 BWR4 STS – Section 3.6.1.4 (Containment Systems))
2. (*Drywell Average Air Temperature* shall be ≤135°F (Applicable to Modes 1–3). (Ref. NUREG 1433 BWR4 STS – Section 3.6.1.5 (Containment Systems))
3. *Suppression Pool Average Temperature* shall be: (a) ≤95°F when any operable intermediate range monitor (IRM) channel is > 25/40 divisions of full scale on range 7 with thermal power >1% RTP, and no testing that adds heat to the suppression pool is being performed, and (b) ≤105°F when any operable IRM channel is > 25/40 divisions of full scale on range 7 with thermal power > 1% RTP, and testing that adds heat to the suppression pool is being performed, and (c) ≤110°F when all operable IRM channels are ≤25/40 divisions of full scale on Range 7 with thermal power ≤1% RTP (applicable to Modes 1–3). (Ref. NUREG 1433 BWR4 STS – Section 3.6.2.1 (Containment Systems))
4. *Suppression Pool Water Level* shall be ≥12 ft 2 inches and ≤12 ft 6 inches (applicable to Modes 1–3). (Ref. NUREG 1433 BWR4 STS – Section 3.6.2.2 (Containment Systems))
5. *Drywell Pressure* shall be maintained ≥1.5 psid above the pressure of the suppression chamber. Applicable in Mode 1 (a) From 24 hours after thermal power is > 15% RTP following startup to (b) 24 hours prior to reducing thermal to < 15% RTP prior to the next scheduled reactor shutdown. (Ref. NUREG 1433 BWR4 STS – Section 3.6.2.5 (Containment Systems))

6. Primary Containment Oxygen Concentration shall be < 4.0 volume percent. Applicable in Mode 1 (a) From 24 hours after thermal power is > 15% RTP following startup to (b) 24 hours prior to reducing thermal to < 15% RTP prior to the next scheduled reactor shutdown. (Ref. NUREG 1433 BWR4 STS – Section 3.6.3.2 (Containment Systems))

### *BWR-6 Plants*

1. *Primary Containment to Secondary Containment Differential Pressure* shall be ≥ -0.1 psid and ≤1.0 psid (applicable to Modes 1–3). (Ref. NUREG 1434 BWR6 STS – Section 3.6.1.4 (Containment Systems))
2. *Primary Containment Average Air Temperature* shall be ≤95°F (applicable to Modes 1–3). (Ref. NUREG 1434 BWR6 STS – Section 3.6.1.5 (Containment Systems))
3. *Suppression Pool Average Temperature* shall be: (a) ≤95°F when any operable intermediate range monitor (IRM) channel is > 25/40 divisions of full scale on range 7 with thermal power >1% RTP, and no testing that adds heat to the suppression pool is being performed, and (b) ≤105°F when any operable IRM channel is > 25/40 divisions of full scale on range 7 with thermal power > 1% RTP, and testing that adds heat to the suppression pool is being performed, and (c) ≤110°F when all operable IRM channels are ≤25/40 divisions of full scale on Range 7 with thermal power ≤1% RTP (applicable to Modes 1–3). (Ref. NUREG 1434 BWR6 STS – Section 3.6.2.1 (Containment Systems))
4. *Suppression Pool Water Level* shall be ≥18 ft 4.5 inches] and ≤18 ft 9.75 inches (applicable to Modes 1–3). (Ref. NUREG 1434 BWR6 STS – Section 3.6.2.2 (Containment Systems))
5. *Drywell-to-Primary Containment Differential Pressure* shall be ≥ -0.26 psid and ≤2.0 psid (applicable to Modes 1–3). (Ref. NUREG 1434 BWR6 STS – Section 3.6.5.4 (Containment Systems))
6. *Drywell Average Air Temperature* shall be ≤135°F (applicable to Modes 1–3). (Ref. NUREG 1434 BWR6 STS – Section 3.6.5.5 (Containment Systems))

### *PWR – Babcock &Wilcox Plants*

1. *Containment Pressure* shall be ≥ -2.0 psig and ≤ +3.0 psig (applicable to Modes 1–4). (Ref. NUREG 1430 B&W STS – Section 3.6.4 (Containment Systems))
2. *Containment Average Air Temperature* shall be ≤130°F (applicable to Modes 1–4). (Ref. NUREG 1430 B&W STS – Section 3.6.5 (Containment Systems))

### *PWR – Combustion Engineering Plants*

1. *Containment Pressure* shall be (Dual) > 14.375 psia and < 27 inches water gauge or (Atmospheric) ≥ -0.3 psig and ≤ +1.5 psig (applicable to Modes 1–4). (Ref. NUREG 1432 CE STS – Section 3.6.4 (Containment Systems))
2. *Containment Average Air Temperature* shall be ≤120°F (applicable to Modes 1–4). (Ref. NUREG 1432 CE STS – Section 3.6.5 (Containment Systems))

### *PWR – Westinghouse Plants*

1. *Containment Pressure* shall be (a) ≥ -0.3 psig and ≤ +1.5 psig (applicable to atmospheric, dual, and ice condenser containments) or (b) *Containment Air Partial Pressure* shall be ≥9.0 psia and within the acceptable operation range shown on Figure 3.6.4B-1 (applicable to subatmospheric containments) (applicable to Modes 1–4). (Ref. NUREG 1431 W STS – Section 3.6.4A/B (Containment Systems))
2. *Containment Average Air Temperature* shall be (a) ≤120°F (applicable to atmospheric and dual containments), (b) (i) ≥85°F and ≤110°F for the containment upper compartment and (ii) ≥100°F and ≤120°F for the containment lower compartment (applicable to ice condenser containments) or (c) ≥86°F and ≤120°F (applicable to subatmospheric containments). (applicable to Modes 1–4). (Ref. NUREG 1431 W STS – Section 3.6.5A/B (Containment Systems))

# B
# COMPENDIUM OF DSA AND PRA CODES

This appendix provides a reference listing of analytical software packages used to perform analyses that are used in RISMC that were identified during this research. The codes are classified into the following groups:

- Risk analysis codes (Table B-1)
- Computational fluid dynamics and thermal hydraulics codes (Table B-2)
- Configuration risk management codes (Table B-3)
- Fission product transport and dose assessment codes (Table B-4)
- Consequence analysis (PRA Level 3) codes (Table B-5)
- Structure analysis/risk assessment codes (Table B-6)
- Data management codes (Table B-7)
- EPRI supplied codes (Table B-8)

For each code, a brief description is provided. The following information also is provided in the table for each code:

- Developer
- Applicable PRA level
- Reference information

**Table B-1**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| ASTEC | IRSN, GRS | ASTEC Req | 2 | The purpose of the ASTEC software package (Accident Source Term Evaluation Code) is to simulate all the phenomena that occur during a severe accident in a light water reactor, from the initiating event to the possible release of radioactive products (the 'source term') outside the containment. |
| ATHLET | Gesellschaft fur Anlagenund Reaktorsicherheit (GRS) | http://www.grs.de/en/forschung_entwicklung/simulationsprogramme/athlet.html?pe_id=94 | 2 | ATHLET (Analysis of THermal-hydraulics of LEaks and Transients) is being developed by GRS for the analysis of the whole spectrum of leaks and transients in PWRs and BWRs. |
| BACFIRE | JBF Associates, Inc. | http://www.nea.fr/abs/html/nesc1020.html | 1 | BACFIRE, designed to aid in common cause failure analysis, searches among the basic events of a minimal cut set of the system logic model for common potential causes of failure. |
| BURD | Korea Atomic Energy Research Institute | http://www.nea.fr/abs/html/nea-1819.html | 1 | BURD (Bayesian Update for Reliability Data) is a simple code that can be used to obtain a Bayesian estimate easily in the data analysis of PSA (Probabilistic Safety Assessment). According to the Bayes' theorem the code facilitates calculations of posterior distribution given the prior and the likelihood (evidence) distributions. |
| CAST3M | CEA | http://net-science.irsn.org/scripts/net-science/publigen/content/templates/show.asp?P=2949&L=EN | 3 | CAST3M is used to model the mechanical integrity of the containment during a severe accident |
| CFAST | National Institute of Standards and Technology | http://code.google.com/p/cfast/ | | CFAST is a computer program that fire investigators, safety officials, engineers, architects and others can use to simulate the impact of past or potential fires and smoke in a specific building environment |
| COMCAN | EG&G Idaho, Inc. | http://www.nea.fr/abs/html/nesc0704.html | 1 | The COMCAN fault tree analysis codes are designed to analyze complex systems, such as nuclear plants for common causes of failure. |
| DORIAN | EG&G Idaho, Inc. | http://www.nea.fr/abs/html/nesc1146.html | 1 | DORIAN is an integrated package for performing Bayesian aging analysis of reliability data; for example, for identifying trends in component failure rates and/or outage durations as a function of time. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| EVNTRE | Sandia Nat Lab | http://www.nea.fr/abs/html/psr-0465.html | 1, 2 | EVENTRE is a generalized event tree processor that was developed for use in PRA for severe accident progression of nuclear plants but can be used to analyze progression of other events leading to critical scenarios as well. |
| FRANTIC-3 | Brookhaven National Laboratory | http://www.nea.fr/abs/html/nesc0766.html | 1 | FRANTIC3 is an accident sequence and event tree analysis code for system availability and operation. FRANTIC3 was developed to evaluate system unreliability using time-dependent techniques. The code provides two major options: to evaluate standby system unavailability or, in addition to the unavailability, to calculate the total system failure probability by including both the unavailability of the system on demand as well as the probability that it will operate for an arbitrary time period following the demand. Time-dependent and test frequency dependent failures, as well as demand stress related failure, test-caused degradation and wear-out, test associated human errors, test deficiencies, test override, unscheduled and scheduled maintenance, component renewal and replacement policies, and test strategies can be prescribed. |
| FTA | Atlantic Richfield Hanford Company & Computer Sciences Corporation | http://www.nea.fr/abs/html/nesc0666.html | 1 | The FTA (Fault Tree Analysis) system was designed to predict probabilities of the modes of failure for complex systems and to graphically present the structure of systems. |
| FTAP | Lawrence Livermore National Laboratory | http://books.google.com/books?id=CAUuWcTmNrQC&pg=PA239&lpg=PA239&dq=ftap+Lawrence%C2%A0Livermore%C2%A0National%C2%A0Laboratory%C2%A0&source=bl&ots=hnxpInrGOs&sig=aaIaZQdGbzslqL_nICZoR8pcth4&hl=en&ei=Cvs7SubtG9-MtgfO59gd&sa=X&oi=book_result&ct=result&resnum=5 | 1,2 | FTAP is a general-purpose program for deriving minimal reliability cut-set and path-set families from the fault tree for a complex system. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| GASFLOW | Forschungszentrum Karlsruhe, Los Alamos National Laboratory | http://www.gasflow.net/ | 2 | GASFLOW is a multicomponent, multiphase 3d CFD software for the simulation of compressible, reactive flows.<br>Users Group: GASFLOW User Group |
| GEM | Idaho National Engineering & Environmental Laboratory | | 1,2,3 | SAPHIRE includes a separate module called the Graphical Evaluation Module (GEM). GEM provides a highly specialized user interface with SAPHIRE which automates the process for evaluating operational events at commercial nuclear power plants |
| GOTHIC | Numerical Applications Incorporated | GOTHIC.pdf, http://lth.web.psi.ch/codes/GOTHIC.htm | 2 | GOTHIC (Generation Of Thermal-Hydraulic Information for Containments) is a general purpose thermal-hydraulics computer code for design, licensing, safety and operation analysis of nuclear power plant containments and other confinement buildings. Applications of GOTHIC include evaluation of containment and containment sub-compartment response to the full spectrum of high energy line breaks within the design basis envelope as described in the Safety Assessment Report. |
| IMPORTANCE | Minimal Cut Sets and System Availability from Fault Tree Analysis | http://www.nea.fr/abs/html/nesc0779.html | 1 | Minimal Cut Sets and System Availability from Fault Tree Analysis |
| INTEG INSPEC | Savannah River Laboratory | http://www.nea.fr/abs/html/nesc0590.html | 1,2 | Accident Frequencies and Safety Analysis for Nuclear Power Plant |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| IRRAS | Idaho Nat. Lab | http://www.nea.fr/abs/html/ests0003.html | 1,2 | The code is used to perform event tree and fault tree analysis, define accident sequences and basic event failure data. It is used to solve system and accident fault trees, quantify cut sets and to perform uncertainty analysis on the results.<br><br>Disadvantage: Older versions (prior to V 4.16) were creating more cut sets than could be stored. |
| KANT | IRSN | ASTEC Req | 2 | A special probabilistic quantification code (KANT) is notably intended to represent and quantify the severe accident event tree, represent and group together accident progression families, evaluate the release levels corresponding to each family, display results and estimate uncertainties in them. |
| KCUT | Korea Atomic Energy Research Institute | http://www.nea.fr/abs/html/nea-1824.html | 1,2 | KCUT is software to generate minimal cut sets for fault trees. |
| PREP/KITT | EG&G Idaho | http://www.nea.fr/abs/html/nesc0528.html | 1 | The PREP/KITT computer program package obtains system reliability information from a system fault tree. Exact, time-dependent reliability information is determined for each component and for each minimal cut set or path set. The PREP program obtains minimal cut sets by either direct deterministic testing or by an efficient Monte Carlo algorithm. The minimal path sets are obtained using the Monte Carlo algorithm. The reliability information is obtained by the KITT programs from numerical solution of the simple integral balance equations of kinetic tree theory. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| MC3D | IRSN | http://net-science.irsn.org/scripts/net-science/publigen/content/templates/show.asp?L=EN&P=3830 | 2 | MC3D is a general purpose thermo-hydraulic multiphase flow code. The IRSN currently uses it to model steam explosion following fuel-coolant interaction in nuclear reactors (principally water reactors), possible in the event of a severe reactor accident with fuel meltdown. The current standard version of MC3D offers two applications.<br><br>PREMIX is the first of these and is dedicated to studying the fuel-coolant mixing phase. As a general purpose application, it is also used to study other aspects for example, to assess the phenomenon of direct cladding heating. The second application, known as EXPLO, models the explosion propagation phase. With its modular structure, it has also been used for a wide range of specific applications such as sodium fires in fast neutrons reactors and modeling the porous media transfer kinetics of long term nuclear waste storage.<br><br>Advantage: A global tool capable of processing complex multiphase flows. |
| MOCUS | EG&G Idaho | http://www.nea.fr/abs/html/nesc0653.html | 1 | MOCUS used for minimal cut-sets and minimal path-sets from fault tree analysis. |
| MULTIPLET | United Kingdom Atomic Energy Authority | http://www.nea.fr/abs/html/nea-1041.html | 1,2 | Large Event Trees for Risk Assessment Calculation. MULTIPLET calculates the products of conditional probabilities along all the branches of an event tree and performs various analyses of the results. The program is able to perform sensitivity studies. |
| POISSON | Kernforschungszentrum Karlsruhe | http://www.nea.fr/abs/html/nea-1058.html | 1 | POISSON provides analysis solution of poisson problems in probabilistic risk assessment. It uses Bayesian approach with conjugate stage-1 prior and is improved with experience from similar systems to yield stage-2 prior, and the likelihood function from experience with system under study. |
| PRISIM | JBF Associates, Tennessee | http://www.nea.fr/abs/html/ccc-0574.html | 1 | PRISIM allows inspectors to quickly access probabilistic risk analysis (PRA) information and use it to update risk analysis results, reflecting a nuclear plant's status at any time. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| PROSA-1<br>PROSA-2 | Argonne National Laboratory | http://www.nea.fr/abs/html/nesc0778.html | 1 | PROSA2 is an implementation of the probalistic response surface technique developed for use with accident analysis programs to conduct probability studies of hypothetical accidents |
| PSAPACK | IAEA, Vienna | http://www.nea.fr/abs/html/iaea1174.html | 1 | PSAPACK is an integrated fault tree, event tree software package. It links minimum cut sets (MCS) from system fault trees and performs the Boolean reduction. It can also retrieve data from the reliability data base to perform the quantification of accident sequences.<br><br>Advantage: A reliability data base (RDB) module can create a data base retrieving data from a RDB compiled by the IAEA from 21 sources and which contains about 1000 records. Reliability data can also be screened and modified by the user. |
| RAS | EG&G Idaho | http://www.nea.fr/abs/html/nesc0889.html | 1 | The Reliability Analysis Sys-tem, RAS, is an integrated package of computer programs for quantifying fault trees and computing minimal cut-sets for common cause failures. |
| RELAP/<br>REFLA | Japan Atomic Energy Research Institute, Nuclear Power Engineering Test Center-Tokyo | http://www.nea.fr/abs/html/nea-0821.html | 2 | The code was developed to describe the behavior of the reflooding core during the postulated loss-of-coolant accident of pressurized water reactors. It calculates fluid conditions in the system such as flow, pressure, mass inventory, and quality; thermal conditions in the core such as surface temperatures, quench front propagation and heat fluxes. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| RELAP5 | Idaho National Lab | http://www.nea.fr/abs/html/nesc0917.html | 2 | RELAP5 was developed to describe the behavior of a light water reactor (LWR) subjected to postulated transients such as loss of coolant from large or small pipe breaks, pump failures, and so on. RELAP5 calculates fluid conditions such as velocities, pressures, densities, qualities, temperatures; thermal conditions such as surface temperatures, temperature distributions, heat fluxes; pump conditions; trip conditions; reactor power and reactivity from point reactor kinetics; and control system variables.<br><br>Advantage: Automatic time-step control is used for hydrodynamic advancement. RELAP5/MOD1 uses a five equation two-phase flow hydrodynamic model consisting of the two phasic continuity equations, the two phasic momentum equations, and an overall energy equation augmented by the requirement that one of the phases is assumed saturated. In this model only two interphase constitutive relations are required, those for interphase drag and interphase mass exchange.<br><br>IRUG |
| RELAP-UK | United Kingdom Atomic Energy Authority | http://www.nea.fr/html/dbprog/cpsabsabc.html | 2 | The program calculates 2-phase flow thermal- hydraulic transients in water-cooled nuclear reactors by solving approximations to the one-dimensional equations of hydraulics in an arbitrarily connected system of nodes. A one-dimensional fuel pin heat conduction model is incorporated, and pumps, heat exchangers, valves, fill systems and leaks may be represented. A point neutron kinetics model, with Doppler and void feedback, is included. A steady state option is available.<br><br>Advantage: RELAP-UK/MK4 is the latest UK code in the RELAP series. It is the first version with full capability for PWR blowdown. The major improvements over earlier versions are a drift flux model, the Bryce flow-dependent slip correlation, a revised bubble rise model and a generalized "heat slab" option. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| Relex | NASA | JCL Ch 7 | 1 | The Relex software is a risk analysis system developed in collaboration with the National Aeronautical and Space Administration (NASA), with a focus on human factors analysis (HFA). |
| RETRAN | EPRI | Westinghouse.pdf | 2 | RETRAN-02 is the reactor coolant system (RCS) transient analysis computer code. It has point kinetics capability as well as the ability to model two phase flow and detailed steam generator geometry. The NRC has reviewed and approved non-LOCA safety analysis methods that use Westinghouse's RETRAN model.<br><br>Users Group: RETRAN User Group |
| RISKMAN | PLG Inc. | JCL Ch 7 | 1 | RISKMAN performs integrated risk calculations, with interface to other PRA codes including SETS and CAFTA. In addition to perform FT & ET the code also offers the capability to account for multi-branch event trees, external events, and dependencies between systems. |
| RiskSpectrum | RELCON AB, Sweden | RiskSpetrum.pdf | 1, 2 | A minimal cut set (MCS) code widely used in PRA.<br><br>Advantage: Large and complex model processing feasible on PC platforms. |
| SAFE-D/SAFE-R | RSICC | http://www-rsicc.ornl.gov/rsiccnew/cfdocs/qryOnePackage.cfm?Pack_name=SAFE-D%2FSAFE-R&Pack_id=P00496%20&Pack_cpu=MNYCP%20%20%20%20%20%20%20%20%20%20%20&Pack_vers=00%20%20%20%20&Packtitle=Code%20System%20for%20the%20Analysis%20of%20Component%20Failure%20Data%20with%20a%20Compound%20Statistical%20Model. | 1 | Code System for the Analysis of Component Failure Data with a Compound Statistical Model. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| SAFTAC | United Nuclear Industries | http://www.nea.fr/abs/html/nesc0674.html | 1 | SAFTAC is a monte-carlo fault tree simulation tool for system design performance and optimization. The program views the system represented by the fault tree as a statistical assembly of independent basic input events, each characterized by an exponential failure distribution and, if used, a constant or normal repair distribution. |
| SAPHIRE | Idaho National Engineering & Environmental Laboratory | http://www.nea.fr/abs/html/psr-0405.html | 1,2 | SAPHIRE is a collection of programs developed for the purpose of performing those functions necessary to create and analyze a complete Probabilistic Risk Assessment (PRA) primarily for nuclear power plants. The programs included in this suite are the Integrated Reliability and Risk Analysis System (IRRAS), the System Analysis and Risk Assessment (SARA) system, the Models And Results Database (MAR-D) system, and the Fault tree, Event tree and P&ID (FEP) editors.<br><br>Users Group: Saphire Users Group |
| SETS | Sandia National Laboratories | | 1 | SETS is used for symbolic manipulation of set (or Boolean) equations, particularly the reduction of set equations by the application of set identities. It is a flexible and efficient tool for performing probabilistic risk analysis (PRA), vital area analysis, and common cause analysis. |
| SIGPI | Lawrence Livermore National Laboratory | http://www.nea.fr/abs/html/nesc1082.html | 1,2 | SIGPI computes the probabilistic performance of complex systems by combining cut set or other binary product data with probability information on each basic event. |
| SOLOMON | ANSWERS Software Service, U.K. | http://www.sercoassurance.com/answers/resource/areas/software/solomon.php | 1, 2 | SOLOMON provides a structured approach to managing the complex models needed for very large event trees, bringing flexibility to minimize the calculational and administrative overhead.<br><br>Advantage: Introduces the use of supernodes that reduces time taken to evaluate results. Allows rapid assessment of phenomenological and system uncertainties using automatically-generated importance measures. |

**Table B-1 (continued)**
**Risk Analysis Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| STRADE | Available through Radiation Safety Information Computational Center, Oak Ridge | | 1 | STRADE generates matrices of experimental designs based on the Latin Hypercube Sampling (LHS) technique, which can be applied to any kind of sensitivity analysis or system identification problem involving a large number of input variables. |
| TEMAC | Sandia National Laboratories | http://www.nea.fr/abs/html/nesc1084.html | 1 | TEMAC is designed to permit the user to easily estimate risk and to perform sensitivity and uncertainty analyses with a Boolean expression such as produced by the SETS computer program. Sensitivity and uncertainty analyses associated with top events involve mathematical operations on the corresponding Boolean expression for the top event, as well as repeated evaluations of the top event in a Monte Carlo fashion. |
| TONUS | IRSN, CEA | http://net-science.irsn.org/scripts/net-science/publigen/content/templates/show.asp?L=EN&P=3454 | 2 | The TONUS a CFD code is a single tool for processing all phenomena relating to the risk of hydrogen (distribution, combustion, detonation) likely to occur in a pressurized water reactor containment in a severe accident. It incorporates both lumped-parameter (LP) and computational fluid dynamics (CFD) formulations. The TONUS CFD code is presently used to support the hydrogen risk assessment for the European Pressurized Reactor (EPR) plant and to investigate the impact of the two-room concept on hydrogen distribution in the EPR containment.<br><br>Supporting Data: MISTRA, TOSQAN and RUT |

**Table B-2**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| ANSYS Fluent | ANSYS | http://www.ansys.com/products/fluid-dynamics/fluent/ | 1, 2 | ANSYS FLUENT is a CFD software that contains the broad physical modeling capabilities needed to model flow, turbulence, heat transfer, and reactions for industrial applications |
| BETAPROB | IRSN | IRSNcodes.pdf | 2 | BETAPROB code has been developed to exhaustively identify potential containment leakage paths and quantify the probability of such leakage |
| CATHARE | CEA, IRSN, EDF, AREVA | http://www-cathare.cea.fr/scripts/home/publigen/content/templates/show.asp?P=173&L=EN | 2 | The Code for Analysis of Thermal Hydraulics during an Accident of Reactor and safety Evaluation (CATHARE) is a system code for PWR safety analysis, accident management, definition of plant operating procedures and for research and development. It is also used to quantify conservative analysis margins and for licensing. |
| CENTS | Westinghouse Electric Company | CENTS.pdf | 1,3 | CENTS is a simulation tool for analyzing nuclear power plant transients and accidents postulated to occur in plants incorporating CE and Westinghouse NSSS designs. It uses detailed thermal hydraulic modeling of the RCS and SGs. CENTS has the capability to simulate operator actions in order to quantify their impact on plant behavior. It is used to perform a wide spectrum of licensing analyses of events, such as steam line break and rod ejection accidents.<br><br>Disadvantage: Does not explicitly model containment performance. |

**Table B-2 (continued)**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| CFX4 | AEA Technology Engineering Software | http://www.softscout.com/software/Engineering/Computational-Fluid-Dynamics-CFD/CFX-4.html | 1, 2 3? | A CFD code, CFX-4 offers a unique blend of powerful tools including direct CAD access, highly automated geometry creation tools, sliding meshes, and advanced models for turbulence, combustion, radiation, and multiphase flows.<br><br>Advantage: Through the complete understanding of process operations, maximum interfacial area can be achieved in gas-lift reactors, uniform mixing in fluidized beds and enhanced separation in cyclones and settling tanks., CFX-4 offers the most robust and comprehensive set of tools to simulate chemical reactions in liquid, gaseous or multiphase flows and combustion of arbitrary fuels. With these tools combustion efficiency and pollutant emissions can be predicted, fire safety assessments can be performed and mixing sensitive chemistry understood. CFX-4 provides powerful methods for solving large-scale calculations. The parallelization option automatically decomposes the calculation into pieces that fit onto individual processors. The solution then proceeds in parallel, allowing large problems to be computed faster or cases that exceed the capacity of individual processors can be tackled. |
| CIGALON | IRSN | IRSNcodes.pdf | 2 | Models the steam explosion phenomenon produced by contact between the corium (mixture of molten core material and steel) and water present at the bottom of the vessel.<br><br>Supporting Data: BERDA (FZK) |
| COBRA-EN | ENEL SpA, Milan | http://www.nea.fr/abs/html/nea-1614.html, http://www-rsicc.ornl.gov/codes/psr/psr5/psr-507.html | | Starting from a steady-state condition in a LWR core or fuel element, the code allows simulating the thermal-hydraulic transient response to user-supplied changes of the total power, of the outlet pressure and of the inlet enthalpy and mass flowrate. |
| COCOSYS | Gesellschaft fur Anlagenund Reaktorsicherheit (GRS) | http://www.grs.de/en/forschung_entwicklung/simulationsprogramme/cocosys.html?pe_id=166 | 1,2 | COCOSYS is being developed and validated for the comprehensive simulation of severe accident propagation in containments of light water reactors. This system is to allow the simulation of all relevant phenomena, containment systems and conditions during the course of design basis accidents and severe accidents. |

**Table B-2 (continued)**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| CONTAIN | Sandia National Laboratory | NRC Codes Report | 1,2 | CONTAIN is an integral containment analysis code. Uses detailed mechanistic models for predicting the physical, chemical, and radiological conditions inside the containment and connected buildings of a nuclear reactor in the event of an accident. (CONTAIN severe accident model development was terminated in the mid-1990s.) The MELCOR code has similar containment capabilities (but less detailed in some areas) and should generally be used instead of CONTAIN |
| FACTRAN | Westinghouse | Westinghouse.pdf | 1 | FACTRAN is a digital computer code that calculates the transient temperature distribution in a cross section of a metal-clad UO2 fuel rod and the transient heat flux at the surface of the clad, using the nuclear power and time-dependent coolant parameters (pressure, flow, temperature and density) as input. |
| FLOWNEX | SimTec | http://www.simtec.gr/index.php?option=com_content&task=view&id=47&Itemid=37 | 1, 2 | Flownex is a systems CFD code that enables users to perform detail design, analysis and optimization of a wide range of thermal-fluid systems |
| ICARE/ CATHARE | IRSN | http://net-science.irsn.org/scripts/net-science/publigen/content/templates/show.asp?L=EN&P=2820 | 2 | ICARE/CATHARE is composed of a series of modules that each deal with a specific phenomenon: thermohydraulics, thermics, mechanics, chemistry, fission products, movement of materials, debris beds, core meltdown, and so on.<br><br>Supporting Data: PHEBUS, TMI-2 |
| IFCI | Sandia National Laboratory | IFCI.pdf | 1,2 | Integral Fuel-Coolant Interactions (IFCI) code is designed to model the mixing of molten nuclear reactor materials with reactor coolant (water).<br><br>Supporting Data: MAGICO-701, MIXA-6 |
| JERICHO (ESCADRE) | IRSN | IRSNcodes.pdf | 2 | The code has been used to model thermalhydraulic developments and slow hydrogen combustion in the containment. |
| LOFTRAN (including LOFT4) | Westinghouse | WestinghouseTH.pdf | 1 | The LOFTRAN thermal-hydraulic model is best suited for use in transients in which the primary coolant system remains subcooled. The model may also be used for a main steamline break analyses, where two-phase conditions occur in the upper reactor vessel head. |

**Table B-2 (continued)**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| MAAP | Fauske and Associates | http://www.fauske.com/maap.html | 1,2 | The Modular Accident Analysis Program (MAAP) is an integral systems analysis code for assessing off-normal transients that can progress to and include severe accidents.<br>Advantage: Well suited for industry requirements.<br>Disadvantage: Cannot perform as detailed modeling as SCADAP/RELAP5 and so on.<br>Users Group: MUG |
| MELCOR | Sandia National Laboratory | ASTEC Req | 1,2 | MELCOR (Methods for Estimation of Leakages and Consequences of Releases) is a fully integrated computer code that models progression of severe accidents in LWR including thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. |
| NOTRUMP | Westinghouse | WestinghouseTH.pdf | 1 | The NOTRUMP code models one-dimensional thermal-hydraulics using control volumes that are interconnected by flow paths (links). The spacial and time-dependant solution is governed by the integral forms of the conservation equations in the control volumes and flow links. The reactivity feedback is modeled with point kinetics neutronics. |
| OPTOAX | Westinghouse | Westinghouse.pdf | 1 | The OPTOAX code is used to calculate overtemperature delta-T and overpower delta-T reactor trip setpoints. |
| RALOC | Gesellschaft fur Anlagenund Reaktorsicherheit (GRS) | IRSNcodes.pdf | 2 | Performs fast hydrogen combustion. |

**Table B-2 (continued)**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| RELAP5 | INL | http://www.inl.gov/relap5/ | | RELAP5-3D is the latest in the RELAP5 code series developed at the Idaho National Laboratory (INL) for the analysis of transients and accidents in water-cooled nuclear power plants and related systems as well as the analysis of advanced reactor designs.<br><br>Users Group: IRUG |
| RUPUICUV (ESCADRE/ ASTEC) | IRSN | IRSNcodes.pdf | 2 | Vessel failure at high pressure may cause large scale dispersion of corium fragments through the containment, resulting in “direct containment heating” modeled using the code. |
| SCDAP/RELAP5-3D | INL | NRC Codes Report | | SCDAP/RELAP5-3D is the latest in a series of software packages developed at the Idaho National Laboratory (INL) for the analysis of severe accidents in water-cooled nuclear power plants and related systems. The calculations of damage progression include meltdown of fuel rods and structures, fragmentation of embrittled fuel rods, convective and radiative heat transfer in porous debris, formation of a molten pool of core material, and the slumping of material to the lower head.<br><br>Users Group: NRC, D.O.E and others |
| SRAR-CD | CD-adapco | http://www.anl.gov/TRACC/Computing_Resources/star-cd.html | 1, 2 | Multi-physics high-performance computational fluid dynamics (CFD) environment.<br><br>Advantage: STAR-CD is well suited to the solution of large-model simulations that benefit from efficient solution algorithms, memory utilization, and scalability on multiple processors |
| STAR-CCM+ | CD-adapco | http://www.anl.gov/TRACC/Computing_Resources/star-cd.html | 1, 2 | Single Integrated CFD Environment. It uses the well established CFD solver technologies available in STAR-CD, it employs a new client-server architecture and object oriented user interface to provide a highly integrated and powerful CFD analysis environment to users. |

**Table B-2 (continued)**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| THINC IV | Westinghouse | THINC.pdf | 1 | THINC-IV is a three dimensional thermal hydraulic computer program which calculates the local coolant density, mass velocity enthalpy, vapor void, and static pressure distribution in a perimental data verified the calculation method and demonstrated the suitability of THINC-IV for PWR core design analysis. The first subchannel code that included lateral momentum equations to calculate cross flow in rod bundle arrays. |
| TRAC | Los Alamos National Laboratory | http://www.nea.fr/abs/html/nesc0836.html | 1,2 | TRAC is a thermohydraulics, reactor kinetics, 2 phase flow LOCA analysis. TRAC-PF1 performs best-estimate analyses of loss-of-coolant accidents and other transients in pressurized light water reactors. Models employed include reflood, multi-dimensional two-phase flow, nonequilibrium thermodynamics, generalized heat transfer, and reactor kinetics. Automatic steady-state and dump/ restart capabilities are provided. |
| TRAC-BD1 | EG&G Idaho | http://www.nea.fr/abs/html/nesc1031.html | 1,2 | LOCA Analysis of BWR with 3-D Pressure Vessel and Multi Bundle Fuel Model. TRAC-BD1 performs best estimate analyses of loss-of-coolant accidents (LOCA) and other transients in boiling water reactors (BWRs). |
| TRACE | Los Alamos National Laboratory | NRC Codes Report | 1,2 | The TRAC/RELAP Advanced Computational Engine. It analyzes large/small break LOCAs and system transients in both pressurized- and boiling-water reactors (PWRs and BWRs). It simulates postulated accident scenarios to show that the power plant safety systems can bring the plant to safe shut-sown conditions. |
| TRACG | General Electric | http://www.journalarchive.jst.go.jp/english/jnlabstract_en.php?cdjournal=jnst1964&cdvol=35&noissue=8&startpage=607 | 1,2 | TRACG is a new version of the best estimate BWR transient analysis code, which utilizes a multi-dimensional two-fluid model for the thermal hydraulics and a three-dimensional neutron kinetics model. A three-dimensional neutronics, a fully implicit integration scheme and models for advanced BWR components have been implemented in the code upon TRAC-BF1. |

**Table B-2 (continued)**
**Computational Fluid Dynamics (CFD) and Thermal Hydraulic (TH) codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| TRAC-PF1/ EN MOD 3 | ENEL SpA, MILAN | http://www.nea.fr/abs/html/nea-1593.html | 1,2 | Best Estimate Coupled 3-D Neutronics-Thermalhydraulics. TRAC-PF1-EN/MOD3 is a combined computer program comprising a revised version of the TRAC-PF1 transient reactor analysis code and a specially implemented three-dimensional two-group neutron kinetics code (QUANDF). |
| VICTORIA | Sandia National Laboratory | NRC Codes Report | 1,2 | VICTORIA is a radionuclide transport and decommissioning code which provides dose analyses in support of license termination and decommissioning. It is a mechanistic computer code designed to analyze fission product behavior within a nuclear reactor coolant system (RCS) during a severe accident. |
| VULCAIN (ESCADRE) | IRSN | IRSNcodes.pdf | 2 | VULCAIN module used for core degradation (heatup, clad oxidation and hydrogen release, UZr-O interactions, and so on) modeling. |
| WCOBRA/TRAC | Westinghouse | WestinghouseTH.pdf | 1 | WCOBRA/TRAC is a “best-estimate” analysis code used for LBLOCA and long term cooling analyses. |
| WEX | GRS | ASTEC Req | 2 | Corium ablation of the concrete basemat analysis code. It is an improvement of the ASTEC module WECHSL. |
| WGOTHIC | Westinghouse | WestinghouseTH.pdf | 1 | The WGOTHIC additions include a special multi-compartment heat structure component, referred to as the “clime” model, used to model the passive containment cooling system (PCS). The essential features of the PCS include the containment steel shell, the large PCS water storage tank, the weirs on the upper containment dome for flow distribution, and the air flow path consisting of a downcomer, riser, and chimney region. |

**Table B-3**
**Configuration Risk Management Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| ARC | ERIN | http://www.erineng.com/04_Products/arc2.html | 1 | The ARC Workstation provides a complete framework for developing and documenting Appendix R compliance strategy assessments through a logical process of fire scenario evaluation deterministically. |
| ATLAS | ERIN | http://www.erineng.com/04_Products/atlas.htm | 1 | Users navigate through design basis information from the perspective of an object (a specific event, mitigation action, system, component or reference). From the selected perspective, related events, mitigation actions, systems, system functions, components, component functions, engineering requirements, parameters, and references can be viewed. The navigation process is powerful and simple to use and apply. The ATLAS software provides a heuristic environment for Information Management and Knowledge Management. |
| BART | ERIN | http://www.erineng.com/04_Products/bart.html | 1 | The software code to perform Bayesian statistical analysis. The BART software program has been successfully used at several utilities to update PRA component failure rates, initiate event frequencies, common cause factors, and maintenance unavailabilities. |
| COMPAS | ERIN | http://www.erineng.com/04_Products/COMPAS.html | 1 | The software provides a complete electronic device management system for device calibrations, trip tests and data collection for transmission and distribution systems. |
| MARKOV Plus | ERIN | http://www.erineng.com/04_Products/markovplus.html | 1 | The software supports risk-informed in-service inspections (RISI) of piping. The MARKOVPlus workstation interactions between degradation mechanisms and the inspection, detection, and repair strategies that can reduce the probability that failures occur or that failures will progress to ruptures. |
| ORAM/ Sentinel | EPRI (developed by ERIN) | https://www.fbo.gov/index?print_preview=1&s=opportunity&mode=form&id=c885341ad8d00c88b21f727ddf35f726&tab=core&tabmode=list&cck=1&au=&ck= | 1 | ORAM-Sentinel provides a comprehensive environment for performing safety assessments of all plant configurations. The analysis compares unavailability of systems important with safety with plant-specific requirements.<br><br>Disadvantage: No capability to directly solve or manipulate risk models. |

**Table B-3 (continued)**
**Configuration Risk Management Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| PARAGON | ERIN | PARAGON_BriefDescription.doc | | The PARAGON software is a multi-purpose tool intended for use by nuclear plant PRA and non-PRA personnel that provides users a comprehensive ability to implement Configuration Risk Management (CRM) programs to comply with 10CFR 50.65(a)(4), for all modes of operation, using a blended approach to risk assessments. Plant-specific PRA models (for example, CAFTA, WinNupra, RiskSpectrum) can be quantified using PARAGON, and the resulting values for the PRA end-states and importance measures are displayed to the user. The software also provides the ability to combine these quantitative results with deterministic and other qualitative information. |
| PLANTFORMA | ERIN | http://www.erineng.com/04_Products/Applications1.htm | 1 | The program is an integrated reliability, availability, and capacity factor analysis tool for power plants. It provides reliability importance measures with respect to the performance indicators that can be evaluated at various levels, including component level, system level, scenario group level, and individual scenario. |
| REBECA | ERIN | https://www.fbo.gov/index?print_preview=1&s=opportunity&mode=form&id=c885341ad8d00c88b21f727ddf35f726&tab=core&tabmode=list&cck=1&au=&ck= | 1,2 | The program is a tool to perform fault tree and level 1 and level 2 event tree analyses in the graphics mode. REBECA allows user interaction while developing, editing and analyzing fault trees and event trees. The code is no longer readily in use. |
| SAFETY MONITOR | Scientech | https://www.fbo.gov/index?print_preview=1&s=opportunity&mode=form&id=c885341ad8d00c88b21f727ddf35f726&tab=core&tabmode=list&cck=1&au=&ck= | 1 | Safety Monitor is a Windows-based interactive tool that allows plant personnel to evaluate the effects of changes in plant configuration on accident risk. The system operates as a multi-user system, with various features to support work scheduling, operations, PRA, and Maintenance Rule personnel.<br><br>Disadvantage: Does not provide defense in depth calculations. |
| WinNUPRA | Scientech | https://www.fbo.gov/index?print_preview=1&s=opportunity&mode=form&id=c885341ad8d00c88b21f727ddf35f726&tab=core&tabmode=list&cck=1&au=&ck= | 1 | WinNUPRA is a self-contained, integrated, probabilistic safety/risk assessment software package for Level 1 PSA analysis. The flexibility provided by the combined event tree/fault tree. Approach makes WinNUPRA easily adaptable to a wide range of applications.<br><br>Users Group: NUPRA User's Group (NUG) |

**Table B-4**
**Fission Product Transport and Dose Assessment Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| CESAR (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | Models thermal-hydraulics behavior in RCS. |
| CPA (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | Some aerosols are deposited on containment wall surfaces after concrete ablation. Combustion of hydrogen accumulating in the containment is then possible and may cause dynamic loading of the containment walls. |
| DIVA (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | Diva is used to model core degradation. The core degrades (DIVA module) with steam-induced exothermic oxidation of the zircaloy fuel cladding and associated hydrogen production; molten core material (termed “corium”) then forms at high temperature (up to 3000 °C) and flows down through the core. |
| ELSA (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | Fission products released from degraded fuel rods such as fission gases as well as actinides. |
| IMOD | AECL Canada | IMOD&LIRIC.pdf | 2,3 | Code for containment iodine behavior models for safety analysis. |
| IMPAIR3 | PSI Switzerland | http://sacre.web.psi.ch/codes/main-frames/impair3-main1.htm | 2,3 | Iodine Severe Accident Code (IMPAIR 3) was written to analyze the iodine chemistry in multi-LWR compartments and is an extended and improved version of IMPAIR 2/M. It is not applicable to high temperature chemistry which one would expect to take place in a core region and in the primary circuit.<br><br>Supporting Data: Phebus FP project |
| IODE (ASTEC) | IRSN France | ASTEC.pdf | 3 | Passive coupling or integrated use of important empirical models together with the containment codes provide best estimate treatment of iodine behavior in conjunction with the affecting thermal-hydraulic and aerosol parameters in the aqueous and gas phases as well as the transfer of volatile species between them. |
| IODES | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/ccc-0365.html | 3 | IODES is a dynamic linear compartment model of the global iodine cycle which estimates long-term doses and dose commitments to the world population from releases of I-129 to the environment. |
| ISODOP (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | Describes the behavior and transport of residual heat, along with the activity associated with fission products in the reactor. |

**Table B-4 (continued)**
**Fission Product Transport and Dose Assessment Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| LIRIC | AECL Canada | IMOD&LIRIC.pdf | 2,3 | A comprehensive mechanistic model, based on our extensive knowledge of relevant chemical reactions and mass transport.<br><br>Disadvantage: Due to its complexity and size, integration of LIRIC into a safety analysis code is considered to be impractical |
| MATADOR | Battelle Memorial Institute | http://www-rsicc.ornl.gov/codes/ccc/ccc6/ccc-689.html | 2 | MATADOR analyzes the transport and deposition of radionuclides as vapor or aerosol through Light Water Reactor (LWR) containments during severe accidents and calculates environmental release fractions of radionuclides as a function of time. It is intended for use in system risk studies. The principal output is information on the timing and magnitude of radionuclide releases to the environment as a result of severely degraded core accidents. MATADOR considers the transport of radionuclides through the containment and their removal by natural deposition and the operation of engineered safety systems such as sprays. Input data on the source term from the primary system, the containment geometry, and thermal-hydraulic conditions are required. |
| PSIODINE | PSI, Switzerland | PSIodine.pdf | 3 | Models most foreseeable radiochemical reactions involved and are generally used for detailed studies and thorough understanding.<br><br>Users Group: RADTRAD Industry Users Group |
| SOPHAEROS (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | Models aerosols and fission product transport in RCS.<br><br>Supporting Data: PHEBUS, FALCON, TRANSAT, AERODEVAP, DEVAP, TUBA |
| SYSINT (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | SYSINT is used in the management of safeguard systems such as containment spray or the accumulator tanks |
| TRAP-MELT | Battelle | TRAP-MELT.pdf | 2 | Models aerosol and fission product transport behavior. |
| WEX (ASTEC) | IRSN, GRS | ASTEC.pdf | 2 | WEX models ablation of the concrete layer and release of noncondensable gases (H2, CO, CO2, and so on) in the containment. |

**Table B-5**
**Consequence Analysis (PRA Level 3) Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| ALOHA | DOE | http://www.hss.energy.gov/nuclearsafety/qa/sqa/central_registry/aloha/aloha.htm | | The Arial Locations of Hazardous Atmospheres (ALOHA) is an atmospheric dispersion model maintained by the Hazardous Materials Division of National Oceanic and Atmospheric Administration (NOAA). ALOHA is one of three separate, integrated software applications in the Computer-Aided Management of Emergency Operations (CAMEO) suite. |
| CAP-88 | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/ccc-0542.html | 3 | Dose Risk Assessment from Air Emissions of Radionuclides. CAP-88 estimates health impacts from the inhalation, ingestion, air immersion and ground surface irradiation pathways, and tabulates results for maximally exposed individuals and regional populations out to 80 kilometers.<br><br>Disadvantage: The model is intended only for evaluating low-level, chronic releases; it is not appropriate for short-term accidental releases. |
| CRAC2 | Sandia National Laboratories | http://www.nea.fr/abs/html/ccc-0419.html | 3 | CRAC (Calculation of Reactor Accident Consequences) program developed in support of the Reactor Safety Study, WASH-1400, to assess the risk from potential accidents at nuclear power plants. It (1) models the meteorological dispersion of the cloud of radioactive material; (2) determines the health effects of the material upon the surrounding population; and (3) estimates the costs to the public from the accident. CRAC2 samples specific meteorological conditions from a set of representative reactor locations and probabilistically combines the results to form frequency distributions of consequence from a reactor accident. It requires detailed meteorological, population, economic, and health data. In addition, CRAC2 models emergency planning procedures, such as evacuation. Detailed parametric and sensitivity studies can be simply accomplished in one computer run. |

**Table B-5 (continued)**
**Consequence Analysis (PRA Level 3) Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| CRRIS | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/ccc-0518.html | 3 | CRRIS consists of eight fully integrated computer codes which calculate environmental transport of atmospheric releases of radionuclides and resulting doses and health risks to individuals or populations. Radionuclides are handled by CRRIS either in terms of the released radionuclides or the exposure radionuclides which consist of both the released nuclides and decay products that build up during environmental transport. Radiologic decay and ingrowth, soil leaching, and transport through the food chain are included in the calculations.<br><br>Disadvantage: ANEMOS is not to be used for short-term or accidental releases. It is appropriate only for chronic releases. MLSOIL will truncate radionuclide chains of length greater than 20 |
| DandD Version 2.1 | Sandia National Laboratories | NRC Code Report, NUREG 5512 V2 | 3 | A code for screening analyses for license termination and decommissioning. The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and justifications of parameter selections documented in Volumes 1 and 3 of NUREG/CR-5512. It is the Monte Carlo version of the previous DandD code that allows a better treatment of uncertainty in parameters. |
| DPCT | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/ests0599.html | 3 | DPCT is a probabilistic-deterministic contaminant code for transport in ground eater. The hybrid deterministic-probabilistic model is used. The hybrid method addresses the fundamental problem of describing the spread of a large number of moving reference particles within a region. Statistical features provide a basis for representing an idealized pattern of motion. |
| GENII | Pacific Northwest National Laboratories | http://www.epa.gov/rpdweb00/assessment/genii.html#download | | The GENII System provides software for calculating radiation dose and risk from radionuclides released to the environment. |
| GENII-NESHAPS | Pacific Northwest National Laboratories | http://www.epa.gov/rpdweb00/assessment/genii.html#download | | The GENII-NESHAPS Edition is specifically designed to help site managers plan and improve compliance with 40 CFR 61, Subparts H and I |

**Table B-5 (continued)**
**Consequence Analysis (PRA Level 3) Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| IMPACTS-BRC2.1 | Sandia National Laboratory | http://www.nea.fr/abs/html/ests0005.html | 3 | The code allows calculations to be made of human exposure to the waste by many pathways and exposure scenarios. |
| MACCS2 | Sandia National Laboratories, Oak Ridge National Laboratory, Idaho National Engineering Laboratory | | | MACCS2 contains simple models with convenient analytical solutions. A MACCS2 calculation consists of three phases: input processing and validation, phenomenological modeling and output processing. The phenomenological models are based mostly on empirical data, and the solutions they entail are usually analytical in nature and computationally straightforward. The modeling phase is subdivided into three modules. ATMOS treats atmospheric transport and dispersion of material and its deposition from the air utilizing a Gaussian plume model with Pasquill-Gifford dispersion parameters. EARLY models consequences of the accident to the surrounding area during an emergency action period. CHRONC considers the long term impact in the period subsequent to the emergency action period.<br><br>Advantage: Allow estimates of consequences of releases of all known radionuclides that may be available in nuclear reactor accidents.<br><br>Disadvantage: The atmospheric model included in the code does not model the impact of terrain effects on atmospheric dispersion. The code also does not model dispersion close to the source (less than 100 meters from the source) or long range dispersion. The economic model included in the code models only the economic cost of mitigative actions. |
| PRASMA | Japan Atomic Energy Research Institute | http://www.nea.fr/abs/html/nea-1352.html | 3 | PRASMA is a system of three programs to select off-site protective action in reactor accidents. The model takes into account several consequences caused by reactor accidents, such as fatality, injury and cost. The first code, IND, calculates individual risks at a distance from a reference nuclear power plant under different types of action. POP then evaluates population risks based on IND output, for a given population density GRAPH prepares graphical representations of IND and POP results. |

**Table B-5 (continued)**
**Consequence Analysis (PRA Level 3) Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| RADRISK | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/ccc-0422.html | 3 | RADRISK estimates radiation dose rates to various human organs from inhalation or ingestion of radioactive materials, and the health effects in a reference population as a result of this exposure. |
| RADTRAD | Alion Science and Technology | http://radtrad.com/index.htm | 3 | The RADTRAD code can be used to assess occupational radiation exposures, typically in the control room; to estimate site boundary doses; and to estimate dose attenuation due to modification of a facility or accident sequence as a function of time. |
| RASCAL 3.0.5 | Athey Consulting, West Virginia | http://www.nea.fr/abs/html/nea-0745.html | | Radiological Doses from Accidental Release to Atmosphere. It evaluates releases from nuclear power plants, spent fuel storage pools and casks, fuel cycle facilities, and radioactive material handling facilities (developed for NRC). The source term calculations performed that pertain to fuel-cycle facility and materials accidents can be generally categorized as (1) fuel-cycle facility/UF accidents, (2) uranium fires and explosions, (3) criticality accidents, and (4) isotopic releases (for example, transportation, materials). |
| RESRAD 6.0 | Argonne National Laboratory | NRC Code Report | 3 | The RESRAD code applies to the cleanup of sites. The applications was adapted by Argonne National Laboratory (ANL) for NRC regulatory applications for probabilistic dose analysis to demonstrate compliance with the NRC's license termination rule (10 CFR Part 20, Subpart E). |
| RESRAD-BUILD 3.0 | Argonne National Laboratory | NRC Code Report | 3 | The RESRAD-BUILD code applies to the cleanup of buildings and structures. The applications was adapted by Argonne National Laboratory (ANL) for NRC regulatory applications for probabilistic dose analysis to demonstrate compliance with the NRC's license termination rule (10 CFR Part 20, Subpart E). |
| RISKAP | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/ccc-0486.html | 3 | RISKAP estimates risk to a population exposed to radioactivity. Risk is measured in terms of the expected number of premature deaths resulting from radiogenic cancers, the number of years of life lost as a result of these deaths, and the average number of years of life lost per premature death. |

**Table B-5 (continued)**
**Consequence Analysis (PRA Level 3) Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| TIME-2 | Framework from ONWI, F.F.S.M., Technical sub-models | http://www.nea.fr/abs/html/nea-1077.html | 3 | TIME2 simulates the long-term evolution burial facilities for radioactive waste disposal, and forms part of a development program in time-dependent probabilistic risk analysis. |
| TOXRISK | Sandia National Laboratories | http://www.nea.fr/abs/html/nesc9710.html | 3 | TOXRISK is an interactive program developed to aid in the evaluation of nuclear power plant control room habitability in the event of a nearby toxic material release. The program uses a model which is consistent with the approach described in the NRC Regulatory Guide 1.78. |
| TRAP-SCO | Tractionel, Belgium? | http://www.nea.fr/html/dbprog/cpsabsabc.html | 3 | Evaluation of the pressure and temperature history during the short term following a break of a high energy line in order to design the compartment walls and ensure their integrity. Conservation of mass and energy are solved in a finite difference solution using the predictor-corrector method (Simpson integration). |
| VARSKIN 3 | Southwest Research Institute | http://www-rsicc.ornl.gov/codes/ccc/ccc5/ccc-522.html | 3 | VARSKIN is a computer code for assessing skin dose from skin contamination developed to assist compliance with 10 CFR 20.1201(c) which requires licensees to have an approved radiation protection program that includes established protocols for calculating and documenting the dose attributable to radioactive contamination of the skin. |

**Table B-6**
**Structure Analysis/Risk Assessment Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| CARES | Brookhaven National Laboratory | http://www.nea.fr/abs/html/psr-0388.html | | CARES1.2 is organized in a modular format with the basic modules of the system performing static, seismic, and nonlinear analysis. In this release, only the seismic module is implemented which evaluates the free-field motion, and computes the structural response and floor response spectra including soil-structure interaction. The eight options in CARES1.2 currently are: a general manager for the seismic module, deconvolution analysis, structural data preparation for soil-structure interaction (SSI) analysis, input motion preparation for SSI analysis, SSI analysis, earthquake simulations/data, PSD (Power Spectral Density) related acceleration time history/spectra analysis, and plot generation. |
| CASTEM | CEA | IRSNcodes.pdf | 3 | Performs analysis of containment mechanical behavior |
| FRELIB | University of Liverpool | http://www.nea.fr/abs/html/nea-0692.html | 1 | Performs calculation of the reliability index given the failure boundary. A linearization point (design point) is found on the failure boundary for a stationary reliability index (min) and a stationary failure probability density function along the failure boundary, provided that the basic variables are normally distributed. |
| OCA-P | Oak Ridge National Laboratory | http://www.nea.fr/abs/html/nesc1125.html | 3? | OCA-P is a probabilistic fracture-mechanics code prepared specifically for evaluating the integrity of pressurized-water reactor vessels subjected to overcooling-accident loading conditions. Both deterministic and probabilistic analyses can be performed. For deterministic analysis, it is possible to conduct a search for critical values of the fluence and the nil-ductility reference temperature corresponding to incipient initiation of the initial flaw. The probabilistic portion of OCA-P is based on Monte Carlo techniques, and simulated parameters include fluence, flaw depth, fracture toughness, nil-ductility reference temperature, and concentrations of copper, nickel, and phosphorous. |

**Table B-6 (continued)**
**Structure Analysis/Risk Assessment Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| OCTAVIA | Office of Nuclear Regulatory Research, US NRC | http://www.nea.fr/abs/html/nesc0898.html | 3? | OCTAVIA (Operationally Caused Transients and Vessel Integrity Analysis) calculates the probability of pressure vessel failure from operationally-caused pressure transients which can occur in a pressurized water reactor (PWR). The analysis approach involves calculation of vessel failure pressures using fracture mechanics methods and estimation of pressure-transient characteristics using historical nuclear data. The failure pressure for a given flaw is calculated using linear-elastic and elastic-plastic methods.<br><br>Advantage: Maxima of 100 temperatures, 12 fluences, 8 flaw sizes |
| PASCAL | Japan Atomic Energy Research Institute | http://www.nea.fr/abs/html/nea-1680.html | 3 | PASCAL (PFM analysis of Structural Components in Aging LWR) is a PFM (Probabilistic Fracture Mechanics) code for evaluating the failure probability of aged pressure components of LWR and their structural integrity. Some new analysis models and original methodologies were introduced in PASCAL such as the elastic-plastic fracture criterion based on R6 method, a new crack extension model of semi-elliptical crack evaluation and so on. Moreover a function to evaluate the effect of embrittlement recovery by annealing of irradiated RPV is also introduced in the code based on the USNRC R.G. 1.162(1996). The code has been verified through various failure analysis results and international PTS round robin analysis ICAS which had been organized by the Principal Working Group 3 of OECD/NEA/CSNI. |
| PAVAN | Nuclear Regulatory Commission | http://www.nea.fr/abs/html/ccc-0445.html | 3 | PAVAN estimates down-wind ground-level air concentrations for potential accidental releases of radioactive material from nuclear facilities. |

**Table B-6 (continued)**
**Structure Analysis/Risk Assessment Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| P-CARES 2.0.0 | Brookhaven National Laboratory | http://www.nea.fr/abs/html/psr-0538.html | 1 | P-CARES 2.0.0 (Probabilistic Computer Analysis for Rapid Evaluation of Structures) was developed for NRC staff use to determine the validity and accuracy of the analysis methods used by various utilities for structural safety evaluations of nuclear power plants. P-CARES provides the capability to effectively evaluate the probabilistic seismic response using simplified soil and structural models and to quickly check the validity and/or accuracy of the SSI data received from applicants and licensees. The code is organized in a modular format with the basic modules of the system performing static, seismic, and nonlinear analysis. |
| PC-PRAISE | Lawrence Livermore National Laboratory | http://www.nea.fr/abs/html/ests0071.html | 1 | BWR Piping Reliability Analysis. PC-PRAISE is a probabilistic fracture mechanics computer code to estimate probabilities of leak and break in nuclear power plant cooling piping. The crack growth analysis is based on (deterministic) fracture mechanics principles, in which some of the inputs (such as initial crack size) are considered to be random variables. Monte Carlo simulation, with stratified sampling on initial crack size, is used to generate weldment reliability results. |
| SEISIM-1 | Lawrence Livermore National Laboratory | Seismic Probabilistic Risk Assessment. SEISIM1 calculates the probabilities of seismically induced failures for components and systems and propagates these calculations to determine the probability of accident sequences and the resulting total risk, which is quantified as an expected value of radiation release and exposure from a given nuclear power plant. SEISIM1 requires as input, files created by other programs developed as part of the SSMRP project. In particular, the SMACS program (NESC 9706) provides the response data for SEISIM1. SEISIM1 calls subroutines MDNOR and MDNRIS from proprietary International Mathematical and Statistical Library, Inc. (IMSL); these routines are not included. | | SEISIM1 calculates the probabilities of seismically induced failures for components and systems and propagates these calculations to determine the probability of accident sequences and the resulting total risk, which is quantified as an expected value of radiation release and exposure from a given nuclear power plant. SEISIM1 requires as input, files created by other programs developed as part of the SSMRP project. In particular, the SMACS program (NESC 9706) provides the response data for SEISIM1. SEISIM1 calls subroutines MDNOR and MDNRIS from proprietary International Mathematical and Statistical Library, Inc. (IMSL); these routines are not included.<br><br>Disadvantage: Fewer number of cutsets, basic events, and accident sequences can be analyzed than the typical PRA codes. |

**Table B-6 (continued)**
**Structure Analysis/Risk Assessment Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| SMACS | Lawrence Livermore National Laboratory | | | The SMACS (Seismic Methodology Analysis Chain with Statistics) system of computer programs is one of the major computational tools of the U.S. NRC Seismic Safety Margins Research Program (SSMRP). SMACS is comprised of the core program SMAX, which performs the SSI response analyses, five pre- processing programs, and two postprocessors. SMACS performs repeated deterministic analyses, each analysis simulating an earthquake occurrence. Uncertainty is accounted for by performing many such analyses using different definitions of the seismic input and varying different system parameters according to a Latin hypercube experimental design. SMACS links seismic input with the calculation of soil-structure interaction (SSI), major structure response, and subsystem response. Seismic input is defined by ensembles of acceleration time histories in three orthogonal directions. |
| TORMIS | EPRI | http://adamswebsearch.nrc.gov/scripts/rwisapi.dll/@pip1.env?CQ_SESSION_KEY=LHWVVAXIQYSV&CQ_QUERY_HANDLE=129703&CQQNUM=1&CQ_DOCUMENT=YES&CQ_SAVE[ResultsReturnPage]=results_list.html&CQ_CUR_DOCUMENT=2 | | Tornado Missile Risk Evaluation (TORMIS) computer code methodology employs Monte Carlo techniques to assess the probability that tornado missile strikes will cause unacceptable damage to safety-related plant features developed in response to appendix to Regulatory Guide1.117, “Tornado Design Classification,” Revision 1, issued April 1978, lists the types of SSCs that should be protected from design basis tornadoes. |
| VISA2 | Pacific Northwest Laboratory | http://www.nea.fr/abs/html/nesc1115.html | 2,3 | The solution method uses closed form equations for temperatures, stresses, and stress intensity factors. A polynomial fitting procedure approximates the specified pressure and temperature transient. Failure probabilities are calculated by a Monte Carlo simulation. The deterministic portion of the code performs heat transfer, stress, and fracture mechanics calculations for a vessel subjected to a user-specified temperature and pressure transient. The probabilistic analysis performs a Monte Carlo simulation to estimate the probability of vessel failure. Parameters such as initial crack size and position, copper and nickel content, fluence, and the fracture toughness values for crack initiation and arrest are treated as random variables. |

**Table B-7**
**Data Management Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| FIREDATA | Sandia National Labs | http://www.nea.fr/html/dbprog/cpsabsabc.html | 1 | FIREDATA contains raw fire event data from 1965 through June 1985. These data were obtained from a number of reference sources including the American Nuclear Insurers, Licensee Event Reports, Nuclear Power Experience, Electric Power Research Institute Fire Loss Data and then collated into one database developed in the personal computer database management system, DBASE III. |
| MAPLE | Katholieke Universiteit, Belgium | http://www.nea.fr/abs/html/nea-1096.html | | MAPLE plots fault trees with AND, OR, EOR, majority, NOT gates, primary events and non-developed events (diamonds). Duplicate branches are automatically replaced by input triangles. |
| PRA DocAssist | EPRI | PRA DocAssist Users Manual | | PRA DocAssist is a tool to aid in the documentation of Probabilistic Risk Assessments (PRAs). The numerous word-processed documents used to document PRAs are replaced with a database containing the PRA documentation content. |
| REVELATION | EPRI (developed by ERIN) | http://www.erineng.com/04_Products/revelation.html | 1 | The software translates Probabilistic Safety Assessment (PSA) outputs into user friendly graphs and reports |
| TREDRA | JBF Associates, Inc., Tennessee | http://www.nea.fr/abs/html/nesc1021.html | | TREDRA is a computer program for drafting report-quality fault trees. Output includes fault tree plots containing all standard fault tree logic and event symbols, gate and event labels, and an output description for each event in the fault tree. |

**Table B-8**
**EPRI Supplied Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| CAFTA | EPRI | CAFTA.pdf | 1 | CAFTA is a PC-based fault tree workstation with support for all phases of systems analysis. Includes full screen editor, multilevel reliability database, plotting, cut set generator, cut set results editor. Extensive syntax and logic checking, logical editing, supports macros, calculates unavailability from failure rate and exposure times, user definable fields, truncates on cut set probability or size.<br><br>Users Group: EPRI R&R User Group |
| DPC | EPRI | http://teams.eprisolutions.com/RR/Lists/Tasks/DispForm.aspx?ID=19&Source=http%3A%2F%2Fteams.eprisolutions.com%2FRR%2FLists%2FTasks%2Factive.aspx | 1,2 | Direct Probability Calculator (DPC) is a tool for calculating or estimating an exact top event probability (or frequency) of a fault tree logic model without employing any cutset-based methods by calculating lower and upper bounds.<br><br>Users Group: EPRI R&R User Group |
| EOOS | EPRI | EOOS Help<br>https://www.fbo.gov/index?print_preview=1&s=opportunity&mode=form&id=c885341ad8d00c88b21f727ddf35f726&tab=core&tabmode=list | 1 | EOOS is a computer program for monitoring safety. EOOS is designed for two types of users, each with a distinct set of needs. The first, an Operator, is a user concerned with current plant status. The second user, a Scheduler, is concerned with scheduling future equipment outages.<br><br>Advantage: Fully integrated PSA modeling package to support it EOOS provides the ability to link directly to a number of high performance quantification engines (Le., CAFTA, NUPRA, IRRAS, and so on), and allows the ability to manipulate RISKMAN Models, or cut-set models created by the packages.<br><br>Users Group: EPRI R&R User Group |
| FIVE | EPRI | http://144.58.4.113/PRA/Big%20List%20of%20PRA%20Documents/EPRI%20TR-100370%20(abstract).htm | 1 | Fire-Induced Vulnerability Evaluation (EPRI TR 100370). This report describes the fire-induced vulnerability evaluation (FIVE) methodology, an NRC-approved quantitative screening technique for fire analysis. The methodology is based on conservative assumptions using industrial and plant-specific databases for evaluating fire event sequences, while making maximum use of existing plant fire analyses and documentation.<br><br>Users Group: EPRI R&R User Group |

**Table B-8 (continued)**
**EPRI Supplied Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| FRANX (RR) | EPRI | FRANX.pdf | 1 | FRANX is a personal computer based tool for analyzing fire risk at power plants. It can be used to identify fire zones and the components in each fire zone, locations of cables relative to each zone, fire initiation sources and define fire scenarios, determine the progression of the fire, and from this, identify the equipment in the zones that are damaged to finally calculate the risk of each fire scenario using the PRA model.<br><br>Disadvantage: The actual fire progression modeling is performed by separate software that is not supplied with FRANX.<br><br>Users Group: EPRI R&R User Group |
| HRA Calculator | SCIENTECH, EPRI | HRA Calc Help | 1 | The HRA Calculator is designed to step probabilistic risk assessment (PRA) analysts through the human reliability analysis (HRA) tasks needed to develop and document human failure events (HFEs), and to quantify human error probabilities (HEPs). The HRA Calculator operates on a basic event basis and is based on EPRI's SHARP and SHARP1. The current version of the HRA Calculator applies EPRI's Cause-Based Decision Tree Method (CBDTM), the Human Cognitive Reliability/Operator Reactor Experiments (HCR/ORE), the Accident Sequence Evaluation Program Human Reliability Analysis Procedure (ASEP), the SPAR-H, and the Techniques for Human Error Rate Prediction (THERP). |
| PRAQUANT | EPRI | PRAQuant Help | 1 | PRAQuant is a tool to automate the evaluation process of many fault trees under changing conditions. It can do so by evaluating the cutsets of event tree sequences in order to calculate the final total frequency.<br><br>EPRI R&R User Group |
| Qrecover | EPRI | Qrecover Help | 1 | QRECOVER is a tool to automatically manipulate cutsets based on a set of rules. Typically, this is used to add recoveries to cutsets, although many other types of manipulations are also available.<br><br>Users Group: EPRI R&R User Group |

**Table B-8 (continued)**
**EPRI Supplied Codes**

| Codes | Developer | Reference | PRA Level | Description |
|---|---|---|---|---|
| RBDA | EPRI | | 1 | Reliability Block Diagram Analysis is a graphical depiction of the system's components and connectors which can be used to determine the overall system reliability<br><br>Users Group: EPRI R&R User Group |
| SYSIMP | EPRI | SysImp 2.1 User Manual | 1,2 | SYSIMP (System Importance) is a computer tool that automates the calculation of collective risk importance and helps one visualize the results. SYSIMP can save many work hours of effort in risk models that represent a large number of systems and components.<br><br>Users Group: EPRI R&R User Group |
| UNCERT | EPRI | | 1,2 | This code is used to calculate the uncertainty in parameters such as CDF and LERF. Both the probability distribution function and the cumulative probability distribution function are formulated based on the uncertainty parameters obtained from the type code or basic events file.<br><br>Users Group: EPRI R&R User Group |
| Xinit | EPRI | XInit Presentation | 1 | The XInit External Event Tool can be used to address external event impacts in EOOS. The XInit External Events Tool is easy to use and can quickly modify external event impacts due to changes in plant configurations. Spatial dependency and impact relationships are addressed in MS ACCESS Format<br><br>Users Group: EPRI R&R User Group |

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