APPROACH TO CONSERVATIVE DETERMINISTIC SAFETY ANALYSIS FOR DESIGN BASIS ACCIDENTS BASED ON REGULATORY REQUIREMENTS

N. Mesmous and J. Kowalski
Canadian Nuclear Safety Commission, Ottawa, Ontario, Canada
Noreddine.Mesmous@cnsc-ccsn.gc.ca, Janusz.Kowalski@cnsc-ccsn.gc.ca

Abstract

Canadian reactor licensing practice has evolved along the Limit of Operating Envelope (LOE) approach, in which the plant operating and safety systems parameters are set at their limiting values. In this type of analysis, it is assumed that the conservatisms inherent in setting input plant parameters at their limits will cover the impact of uncertainties associated with the code models. Although LOE methodology provides some conservatism in the analysis results it is not certain if it off-sets the uncertainties associated with modeling parameters relevant to the analyzed event.

A new Conservative Analysis Method (CAM) has been developed that takes into account uncertainties in input modeling and plant parameters when performing conservative safety analysis for design basis accidents. The CAM approach is consistent with the Canadian Nuclear Safety Commission (CNSC) and the International Atomic Energy Agency (IAEA) regulatory requirements. This methodology requires knowledge of key input parameters and the important output parameter(s). Only parameters important for highly ranked phenomena are selected as key input parameters. The selection is performed using the PIRT/PKPIRT processes and confirmed by sensitivity analysis. For each key input parameter, the analysis value for the modeling parameters, the analysis set points for the plant parameters and the actuation of the mitigating system(s) are defined and their impact on the safety analysis results assessed.

An example is presented to show the application of this methodology. The example involves a comparison of the LOE, Best Estimate (BE) and CAM results simulating a small break loss of coolant accident in a generic CANDU 6 station. The calculations are performed with the CATHENA computer code coupled with the LEPCON program.

1. Introduction

The Canadian nuclear industry usually performs safety analysis with best estimate codes and conservative initial and boundary conditions (I&B/C) as well as with conservative assumptions with regard to the availability of systems. This is called limit of operating envelope approach.

It has been argued that LOE assumptions are conservative enough to account for simulation errors. Since uncertainties of the tool used to perform the analysis are generally unknown, it is not clear how the conservatism of the assumptions covers the modeling uncertainties. The LOE methodology may provide more pessimistic results, but cannot provide confidence that the assumed conservatism covers modeling uncertainties of the computer tools used to perform the calculations.

As an alternative to the LOE approach, the Canadian industry has been developing the Best Estimate Analysis and Uncertainty methodology (BEAU) [1] that provides a more realistic quantification of safety margins using the BE tools with an integrated accounting of uncertainties in analysis predictions. The BEAU approach can yield larger safety margins than those obtained with the LOE approach but it is more resource intensive than the other methodologies.

The purpose of this paper is to provide a new technically sound path for taking into account the uncertainties in the key input modeling and plant parameters when performing conservative safety analysis for design basis accidents.

2. Safety analysis requirements

Safety analyses are performed to demonstrate that the mitigating systems in a nuclear power plant can prevent unacceptable consequences and that the acceptance criteria, usually linked to the integrity of barriers preventing the release of radioactive material, are met.

The above two premises can be summarized as follow:

- The safety analysis needs to demonstrate that the mitigating systems can avoid a serious challenge to the barriers that prevent the release of fission products to the environment; and
- The barriers should prevent the release of fission products to the environment. In a CANDU reactor those barriers are: the fuel, the fuel channel (pressure and calandria tubes), the primary heat transport system piping (PHTS) and the containment.

2.1 Canadian regulatory requirements

The Canadian regulatory requirements and recommendations for safety analysis methodologies and tools validation are provided in CNSC RD-310 regulatory document [2], and the Canadian Standard Association CSA N286.7-99 [3].

The CNSC regulatory document requires that a sufficient degree of conservatism should be built in safety analysis to off-set any uncertainties associated with both nuclear power plant (NPP) initial and boundary conditions and modeling of NPP performance in the analyzed event. The safety analysis should also account for the uncertainties in calculations and data to ensure that appropriate margins exist.

The CSA N286.7-99 requires the licensees that the tools used in the safety analysis should be validated and the code accuracy should be the outcome of the validation exercises.

3. Deterministic Safety Analysis Methods

Safety assessment for level three defence in depth of power reactors are mainly performed by deterministic methods that are often characterized by the conservatism. By comparison, the best estimate method provides a realistic simulation of the accident scenarios to a level commensurate with the currently known data and knowledge of the phenomena associated with the events. Different safety analysis methodologies have been developed and used for licensing purposes in Canada. The following section presents some information on the existing Canadian methodologies used in deterministic safety analysis.

3.1 Limit of Operating Envelope (LOE)

In most cases, safety analyses performed by the Canadian nuclear industry use best estimate codes with conservative initial and boundary conditions as well as with conservative assumptions with regard to the availability of systems. The use of conservative assumptions and limiting initial and boundary conditions may result in analyzed plant that states are not realistic. While the LOE analysis methodology could be bounding with respect to operational and safety system parameter values, the same cannot be said for the treatment of the physical models and associated computer codes. In fact, for the most part, best estimate methods are used wherever they are available, and modeling uncertainties are not generally accounted for. If no assessment is made taking into account uncertainties of data and models, it is not clear how it could be shown that the conservatism of the assumptions covers those uncertainties.

3.2 Best Estimate Analysis and Uncertainty methodology (BEAU)

A more realistic estimate of the safety margins can be obtained through the use of best estimate tools supplemented with the evaluation of uncertainties. The approach developed and used in Canada is called Best Estimate Analysis and Uncertainty (BEAU) [1]. This methodology uses BE computer codes with realistic assumptions for the initial and boundary conditions and is complemented by statistical propagation of input uncertainties.

For this approach, the uncertainty analysis should be performed by statistically combining and propagating uncertainties in the input modeling and plant operating parameters in order to establish the bounds of accident consequences with a specified high probability and high confidence.

4. Uncertainties in Deterministic Safety Analysis

The CNSC regulatory document [2] requires considering uncertainties associated with both initial and boundary conditions and modeling of nuclear power plant performance in the analyzed event. A brief description of these uncertainties is provided below.

4.1 Modeling Uncertainties

Prediction errors of computer codes may arise due to various causes, such as the models and numerical solution techniques used, and the accuracy of empirical correlations and of library functions. These factors are grouped under the generic name of modeling uncertainties.

The prediction error has random (or aleatory) and systematic components. The random component is the effect of the randomness of nature that cannot be captured by the model. Systematic errors or bias are usually associated with limitations in the models used to represent the real phenomena. The code accuracy characterized by the bias and the variability of bias, is obtained from the code validation and can be used to determine the input modeling uncertainties.

4.2 Plant Measurement Uncertainties

Plant measurement uncertainty is usually referred to uncertainty in plant parameters that specify the initial operating conditions of the reactor process systems. It is associated with measuring/monitoring of the plant parameters, instrument errors and set points. To determine plant measurement uncertainties, the measurement errors (bias and standard deviation) should be determined and justified. The uncertainty allowance for plant parameters needs to be obtained from operating experience rather than from the values used in the original licensing analysis.

5. New Approach for Implementing Uncertainties in Conservative Safety Analysis

In the LOE safety analysis, a large conservatism is assumed with regard to some plant parameters and availability of the safety systems, however, the above assumptions may not be sufficient to cover uncertainties in the modeling parameters. The BEAU methodology provides more realistic quantification of safety margins with a prescribed probability/confidence level but it requires a large number of code calculations. Based on the knowledge acquired from the above methodologies, an evolutionary and easy to apply approach is proposed.

5.1 Conservative Analysis Methodology (CAM)

In order to achieve confidence in the simulation results, the key input plant and modeling parameters should be considered and their uncertainties evaluated and accounted for.

To ensure that the analysis results are conservative, the value of each key input parameter should take into account the associated uncertainties. In the following sections, a new conservative analysis methodology is presented. It focuses on “what” needs to be done to demonstrate and ensure that the analysis methodology is truly conservative and “how” to take into account the uncertainties in the input key plant and modeling parameters to ensure consistent framework for conservatism in the calculated “output” parameters. Basically, this methodology is evolutionary and is based on lessons learned from BEAU and LOE methodologies. Basic elements of the proposed methodology are shown in figure 1.

The CAM requires knowledge of the important output parameter(s) which generally defines the derived acceptance criteria (Figure(s) of Merit), the key input parameters including the analysis value for the modeling parameters, the set points for the plant parameters and the actuation of the mitigating system(s). Only important parameters for highly ranked phenomena are considered as key input parameters, other parameters are selected at their realistic values.

In a nutshell, the main thrust of this methodology is to efficiently utilize all available types of data to identify important sources of uncertainty, and to assess the magnitude of their impact on the uncertainty of the tool output values.

5.1.1 Figure of Merit (FoM)

FoM is the primary evaluation criterion used to judge the importance of each key phenomenon. It is necessary to ensure that the bounding estimate of the calculated FoM parameter(s) takes into account ***the key input parameters.*** Therefore, the key phenomena and parameters which could have an influence on the results need to be defined.

The deterministic safety analysis should demonstrate that the predicted consequences on the FoM for all event sequences are within the design basis of the nuclear power plant and therefore they satisfy the derived acceptance criteria.

5.1.2 Key input parameters

The phenomena identification and ranking table (PIRT) and phenomena key parameter identification and ranking table (PKPIRT) processes are used to identify and rank key phenomena and accordingly a set of operational and modeling parameters that are potentially important for the assessment.

The PIRT exercise [4], [5] helps to identify the key phenomena associated with the analyzed event which have a significant impact on the FoM. These phenomena are ranked relative to their importance and the safety concern of a plant following the initiation of an accident.

The PKPIRT process [1] identifies and ranks the modeling and plant parameters for all relevant phenomena. It must be demonstrated that this database used in PKPIRT sufficiently covers the range of conditions that are expected to occur during the analyzed event.

Sensitivity analysis may be also used to determine and rank the important input parameters and assess the impact of their variations on the results (FoM). If the results reveal that the model is insensitive to certain parameters then these parameters are set at their best estimate (realistic) values and ranked as not (less) important. However, if the tests reveal that the model is sensitive, then these parameters are ranked accordingly and set at their required values.

5.2 Accounting for uncertainties

In deterministic safety analysis for level-3 defence in depth, all key uncertainties should be identified and accounted for [2]. To ensure that the analysis results are conservative, key modeling and plant parameters uncertainties are taken into account. Less important parameters are set to their nominal values.

5.2.1 Input modeling parameters

Input modeling parameters are those parameters that are used in models to describe the interaction between phenomena and facilitate computer code calculations of nuclear power plant behavior and are typically embedded in computer code models, formulae or correlations (e.g., for instance, heat transfer coefficients).

In the proposed methodology, modeling parameter uncertainties are incorporated by using the key modeling parameter values (KMPV) based on the modeling parameter BE value (MPBEV) and accounting for the code accuracy (ACC). The non-conservative code bias (NCB) and the variation in bias (VB) of the tool(s) are used in performing the safety analysis.
The key modeling parameter value is defined as:

$$KMPV = MPBEV \pm ACC \quad (1)$$

$$ACC = |NCB| + 2\,VB \quad (2)$$

where:

MPBEV is the tool realistic modeling value by which the tool has been validated for the specific application.

ACC is the degree of closeness of a calculated quantity to its true value. It's composed of two components which are the bias and the variability of bias. In Equation 1, the code accuracy should be preceded by the sign that ensures the pessimistic impact.

NCB is a code bias determined during the validation exercise. It represents, for multiple comparisons, the mean value of the differences between the calculated value by the tool and the corresponding true values that could be obtained with the perfect measuring device. Generally, the true value can be taken as the experimental measured value. The bias is *not conservative* whenever, when taken, it would have a pessimistic impact on the results *(conservative results)*.

VB is determined during the validation exercise. It represents the degree of variability in the differences between code calculations and corresponding estimate of true values.

5.2.2 Input plant parameters

Input plant parameters (also referred as operational parameters) are those parameters that characterize the state of nuclear power plant systems or components and can be measured using in-reactor instrumentation (e.g., for instance, reactor coolant temperature).

For key input plant parameters, as they are measured by the plant instrumentation, the total measurement uncertainties should be added (the measurement uncertainty is available from the plant Instrumentation and Control system documentation). In the CAM, the key plant parameter analysis set point (KPPASP) is determined using the measured plant parameter nominal set point (MPPNSP) and the measurement uncertainties (MU) consisting of the non-conservative measurement bias (MB) and standard deviation (σ) as:

$$KPPASP = MPPNSP \pm MU \quad (3)$$

$$MU = |MB| + 2\,\sigma \quad (4)$$

where:

MPPNSP is the parameter measured mean value recorded by the instrumentation and control systems.

MB denotes the measurement uncertainty arising from a systematic error that is known to cause deviation in a fixed direction.

σ represents an element of measurement uncertainty that cannot be defined exactly or that can cause deviation in either direction but that can be estimated on the basis of a probability distribution.

In Equation 3, the measurement uncertainty should be preceded by the sign that ensures the pessimistic impact. In the performed safety analysis, less important plant parameters should be set to their nominal realistic, centered values.

The key plant parameter analysis set point (KPPASP) can be taken directly from the safe operating envelope (SOE) or the operational limits and conditions (OLCs) as it covers the measurement uncertainties of the parameter.

5.2.3 Actuation of mitigating systems

In conservative safety analysis, it is necessary to demonstrate that the mitigating systems can avoid serious challenges to the barriers that prevent the release of fission products to the environment.

The analysis set point to predict the action of the mitigating (AMSSP) system can be expressed based on the Actuation of Mitigating System Nominal Set Point (AMSNSP) as:

$$AMSSP = AMSNSP \pm MU \quad (5)$$

$$MU = |MB| + 2\sigma \quad (6)$$

where:

AMSNSP is a parameter obtained from the overall design and safety analysis and is used to trigger the mitigating system. In such case, there is no need to include additional prediction uncertainty or code accuracy for AMSNSP. For this reason, only the non conservative bias and standard deviation of the measured plant parameters used to trigger

the action of mitigating systems are accounted for. The AMSSP can be determined directly from the SOE or the OLCs.

In equation 5, the MU should be preceded by the sign that ensures a pessimistic impact.

6. Example of application

To demonstrate the CAM applicability, simulations of small break loss of coolant accident (SLOCA) event were conducted for CANDU-6 end of life conditions using a 2.5% break size. The break was assumed to occur at inlet header #2 after the reactor reached full-power conditions. The break opening time to full size is 0.001s in all the analyzed cases.

The value of the results documented here is limited since this example was only intended to present how to deal with including modeling and plant parameters uncertainties in the calculations.

The calculations were performed with CATHENA 3.5d-Rev2 coupled with the LEPCON program. CATHENA (Canadian Algorithm for THErmalhydraulic Network Analysis) [6] is a thermalhydraulic computer code that models the dynamical evolution of one-dimensional flow of two fluids in mechanical and thermal non-equilibrium in a pipe network. LEPCON is used for emulating the most important plant control routines (logic controlling of emergency core cooling "ECC" system).

Details on assumptions and initial plant conditions used in simulation for LOE, BE and CAM methodologies are provided below.

6.1 CASE 1. LOE safety analysis

For the LOE type analysis, conservatism is assumed in the plant initial and boundary conditions, and component and system responses. Availability and efficiency of the systems credited in analysis are to be at the worst permissible levels. Additional assumptions include:

- some boiler tubes are plugged,
- no credit for the actuation of the most effective shut-off rods,
- unavailability of some ECC valves for ECC injection or boiler crash cool-down,
- conservative initial conditions for the ECC system.

Similar assumptions are used in the SLOCA analysis for the Point Lepreau reactor.

Although the operating power for the Point Lepreau reactor at 100% full power is 2056MW, for the LOE analysis it is increased by 3% to 2118MW (Tables 1, 2). As the inlet header temperature can affect the time of the trip signal on low pressurizer level, the models from those references in the LOE cases are run with initial conditions of 270ºC in the inlet header temperature.

The plant parameters credited for the reactor trip are set at the low Primary Heat Transport (PHT) pressure signal and the low pressurizer level signal. The assumed values are the respective nominal trip set point values, 8.8MPa and 7.26m.

6.2 CASE 2. BE safety analysis

In the BE simulations, all the initial and boundary condition parameters (Table 2) are set at their realistic values. The assumed reactor power is 1809.3MW (end of life ~88%) and reactor inlet header temperature initial condition is set at 267ºC.

The plant parameters credited for the reactor trip are set at the low Primary Heat Transport (PHT) pressure signal and the low pressurizer level signal. The assumed values are the respective nominal trip set point values, 8.8MPa and 7.26m.

6.3 CASE 3. CAM safety analysis

The PIRT/PKPIRT processes are applied to identify and rank a set of operational and modeling parameters that had a significant affect on the FoM and actuation of the mitigating system. The FoM chosen as the evaluation criterion to assess the results is the time occurrence of dry-out.

The results from the above processes show that the reactor power and inlet header temperatures are key operational parameters and the critical heat flux (CHF) and nucleate boiling heat transfer coefficients are key modeling parameters (Table 1).

To confirm the results of the PIRT/PKPIRT analysis, sensitivity studies have been performed using the operational and modeling uncertainties as specified in Table 2. Similar modeling and plant uncertainties were applied for the CAM simulations.

For the input modeling parameters (nucleate boiling heat transfer and critical heat flux), the computer accuracy was set at 10% for each parameter. These uncertainties were chosen arbitrarily because the accuracy of the CATHENA code for the prediction of those parameters was not available.

For CAM simulations, the reactor power (91%) and inlet header temperature were higher by 3% and 2ºC, respectively, compared to their values used in BE simulation (Tables 1, 2).

6.4 Discussion of Analysis Results

Table 3 presents the results of a sensitivity analysis, where the time of dry out and the time of trip on each trip parameter (with and without the measurement uncertainties) have been presented for each case. The analysis shows that an increase of the reactor power by 3% and inlet header temperature by 2ºC delays timing of the reactor trip. On the other hand, reduction of the boiling heat transfer coefficient and CHF by 10% does not show any considerable impact on the

trip times based on the low HTS pressure and on low pressurizer level. For this reason, it can be assumed that only measurement uncertainties of the plant parameters may need to be considered when the conservative values for actuation of the mitigating system are calculated.

Decrease of the boiling heat transfer coefficient and CHF by 10% results in faster occurrence of fuel dryout. Slightly earlier dryout is predicted when the uncertainties in the reactor power and inlet are accounted for. In this case, dryout is predicted to occur from sooner than that obtained from calculations that do not account for uncertainties.

Table 4 shows timing of the reactor trip and occurrence of dryout predicted by CATHENA for the limit of envelope (LOE), the best estimate (BE) and CAM methodologies. The reactor trip time calculated by the LOE method varied from 42.2s (for the trip based on low HTS pressure) to 63.3s (for the trip based on the pressurizer level), and occurs sooner than the onset of dryout (76s). Significant margins between the trip time and the time of dryout are predicted using the BE methodology that does not consider uncertainties. Results show that time of the reactor trip is predicted to occur at 41.9s (trip based on the low HTS pressure) or at 21.1s for a trip signal based on the pressurizer level, and the dryout time at 152s. Although the predicted margins are substantially larger than those obtained by the LOE method, the conservatism of prediction cannot be sustained since it does not take into account any modeling and plant parameter uncertainties.

To demonstrate that the safety analysis results are truly conservative, the CATHENA predictions using the proposed CAM approach have been performed. Since sensitivity analysis shows an insignificant effect of input plant parameters on the time of dryout, no modeling uncertainties are accounted for but the measurement uncertainties of the trip parameters are included in the safety analysis calculations.

As presented in table 4 and figure 2, the uncertainties included in the CAM methodology produce conservative results compared to the BE methodology. The time of dryout obtained using the CAM methodology occurred at 104s and was shorter than that calculated by the BE methodology. However, the reactor trip is delayed when the measurement uncertainties of the trip setpoints are included. The predicted margins defined as a difference between time of the reactor trip and time of dryout decreased from 110.8s for BE method to 44.6s for the CAM method and to 12.3s for LOE method.

Preliminary simulation results presented above indicate that for this specific example, the LOE methodology is conservative enough to compensate the neglecting the modeling uncertainties. The CAM methodology also demonstrates that the mitigating systems can avoid a serious challenge to the barriers preventing a release of fission product to the environment.

It is noted that the modeling uncertainties in the CAM simulations are not obtained from the code accuracy but they are assumed to be 10% of the predicted values. Therefore, the results predicted by the CAM methodology should be treated with some caution until the code accuracy is obtained.

7. Conclusions and Recommendations

A new CAM method has been developed for performing conservative deterministic safety analysis for level three defence in depth of nuclear power plants. The proposed method is simple and technically sound and addresses the impact of input modeling and plant uncertainties on the conservatism in safety analysis.

Evaluation of key input parameters is an important part of the CAM methodology and it can be performed either by PIRT studies or by sensitivity analysis. The above methods identify the key phenomena and parameters relevant to the analyzed accident scenario and assess their impact on the FoM.

To ensure that the safety analysis results are conservative, the CAM methodology accounts for key modelling and input plant parameters uncertainties. Also, uncertainties relevant to actuation of mitigating systems are considered.

To demonstrate the applicability of the CAM methodology, CATHENA simulations for the 2.5% break SLOCA event have been performed. The results show that the inclusion of modeling uncertainties has a pronounced impact of the time of dryout but a negligible impact on the reactor trip times.

Differences between the reactor trip time and occurrence of dryout predicted by CAM methodology are smaller than the BE method without considering uncertainties but larger than for LOE. This indicates that for this specific case, LOE is conservative enough to compensate for the modeling uncertainties included in the CAM methodology.

Conclusions relevant to the CAM methodology should be treated with some caution since the validation of CATHENA has not been completed. Therefore, code accuracy for the modeling parameters cannot be determined yet with confidence. Also, further work is needed to assess the applicability of the CAM method for different accident scenarios.

Acknowledgements

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8. References

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Table 1 Key Parameters from PIRT and Uncertainties used in Sensitivity Analyses

Input parameters	Plant or modeling	Range of the parameter
Reactor Inlet Header Temperature	Plant	BE + 2°C
Plant Initial Power	Plant	BE + 3%
Boiling Heat Transfer	Modeling	BE – 10%
Critical Heat Flux	Modeling	BE – 10%

Table 2 Summary of Initial Conditions

CASE **2.5% RIHB**	Initial Power (MW)	Inlet Header Temperature (IHT) (°C)	Outlet Header Pressure (MPa)	Pressurizer Liquid Level (m)
LOE	2118.0	270.3	10.0	13.3
BE	1809.3	266.8	9.98	8.8
CAM -Plant Parameters				
(BE Power +3%)	1863.6	267	9.98	8.9
(BE IHT + 2C)	1809.3	268.6	9.98	9.1
CAM- Modeling parameters				
(CHF-10%)	1809.3	266.8	9.98	8.8
(HTC - 10%)	1809.3	266.8	9.98	8.8

Table 3 Results of Sensitivity Analysis

CASE **2.5% RIHB**	Rector Trip Time due to Low HTS Pressure (s)	Reactor Trip Time due to Low Pressurizer Level (s)	Time of Dryout (s)
Plant Parameters			
Effect of power uncertainty	45.3	25.6	147.9
Effect of the IHT uncertainty	46.8	29.4	149.7
Modeling parameters			
Effect of uncertainty in CHF	41.9	21.1	141.4
Effect of uncertainty in HTC	41.7	21.0	141.4

Table 4 - CATHENA Simulation Results

Methodology used for safety analysis	Time of Trip due to Low HTS Pressure (s)	Reactor Trip Time due to Low Pressurizer Level (s)	Time of Dryout (s)	Difference between Times of Trip and Dryout (s)
LOE	42.2	63.3	76	12.3
BE	41.9	21.1	152.7	110.8
CAM	59.4	43.8	104	44.6

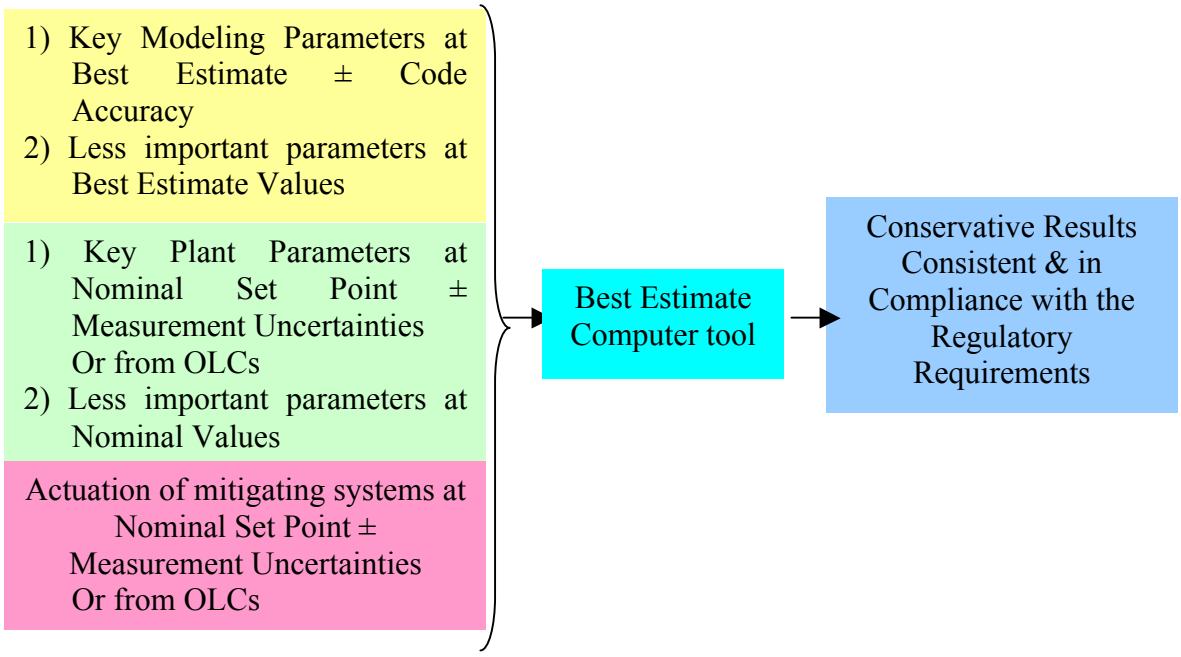

Figure 1 Key Components of New Method for Conservative Safety Analysis

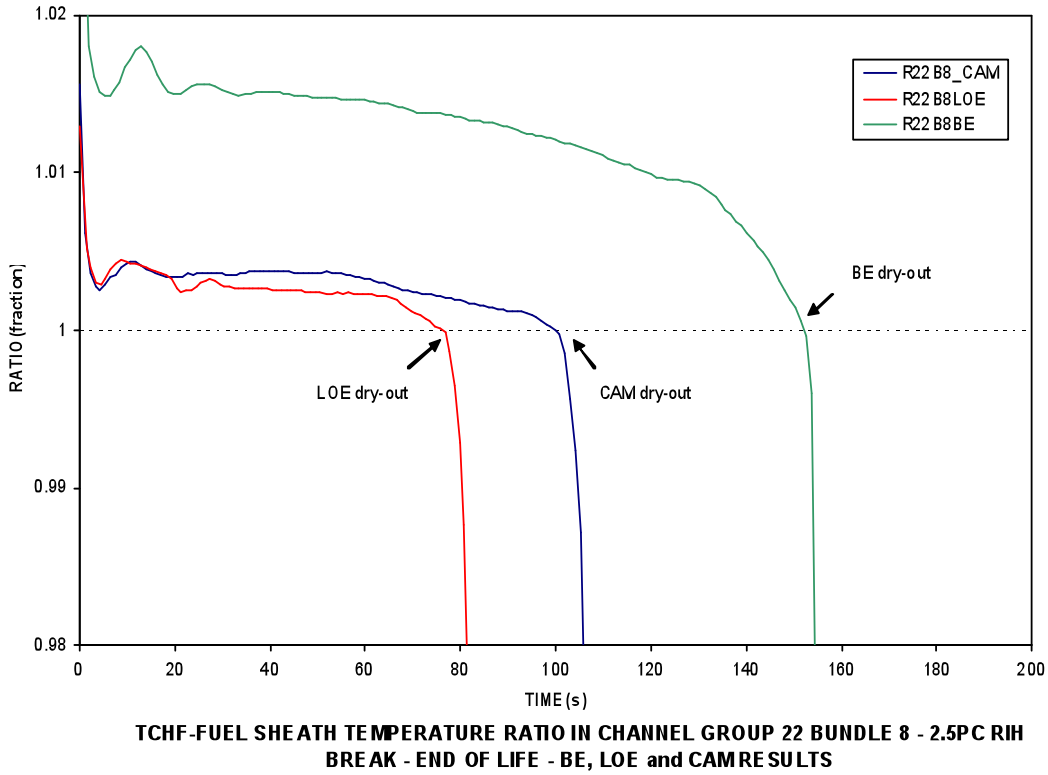

Figure 2 CATHENA Simulation Results for BE, LOE and CAM Methodologies