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Best Estimate Evaluation of Steam Line Break Accident Using Uncertainty Quantification Method

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ABSTRACT

Uncertainty quantification evaluation using a best estimate methodology, the CABUE technique, was applied to the steam line break analysis for UCN 3&4. In the evaluation, uncertainty parameters were identified based on PIRT process and their associated distribution types and ranges were conservatively assumed. Then, the accident simulations were performed by using the RETRAN-3D code for selected uncertainty parameter sets. As a safety parameter reactivity margin to criticality after reactor trip was evaluated. The 95 percentile of the reactivity was determined by Wilks formula from the reactivity values from 59 code simulations with 59 randomly sampled parameter vectors. The resultant reactivity was compared against that of licensing calculation. The comparison showed that applying the CABUE methodology to non-LOCA safety analysis has potential to enhance safety margin. However, the success of the methodology relies on the adequacy of statistical data and identification process associated with the uncertainty parameters. In order to establish more reliable best estimate analysis methodology for non-LOCA transient analysis, more intensive and careful study is necessary on the identification of uncertainty parameters, determination of distribution types and ranges of the parameters, and computer model validation by comparison with plant transient data .

I. INTRODUCTION

As the knowledge advances on the thermal-hydraulic phenomena in nuclear power plants through experimental tests, extensive application of computer resources, and increased operating experiences, there has been an increasing tendency toward a best estimate analysis methodology for safety analysis. However, most researches on the best estimate approach have been mainly focused on the LOCA analysis [1,2]. Recently, several efforts to apply best estimate approaches to non-LOCA analysis have been made [3]. In this paper, an uncertainty quantification evaluation was performed for a typical non-LOCA accident, the steam line break (SLB) accident, using a best estimate methodology, Code Accuracy Based Uncertainty Evaluation (CABUE) technique [4]. The SLB with offsite power available was selected as the accident scenario, and post trip reactivity margin was evaluated as a safety parameter.

The CABUE methodology consists of identification of uncertainty parameters, determination of statistical characteristics of those parameters, and evaluation of uncertainty propagation to the safety parameters. Uncertainties associated with the analysis can be divided into two categories; code uncertainty parameters such as critical flow model and heat transfer correlations implemented in the code, and plant operational and design uncertainty parameters such as core physics, components performances, and operating conditions. An extensive Phenomena Identification and Ranking Table (PIRT) for SLB was performed by a team of experts from research institute, industries, and regulating body to identify a set of important parameters prior to uncertainty analysis [5]. After identifying important uncertainty parameters, their statistical characteristics such as ranges and distributions were determined. Then, the uncertainty propagation of combined effects of individual parameters was evaluated using the Distribution-Free Percentile Estimation (DFPE) technique with Wilks formula [6].

II. ANALYSIS

II-1. Uncertainty Analysis Method for CABUE

The CABUE technique [4] is an uncertainty quantification method newly developed by Korea Electric Power Research Institute (KEPRI) for the application to the safety analysis for Westinghouse 3-loop nuclear power plants. Compared with the Code Scaling Applicability and Uncertainty (CSAU) method [1], which has been a general guide for LOCA uncertainty analysis methodology, CABUE treats the uncertainties associated with the physical models implemented in the simulation code in a more reliable manner. Basic mathematical basis for uncertainty quantification method for CABUE is briefly described below.

The prime objective of the CABUE technique is to develop an uncertainty extrapolation method. This extrapolation should be based on the code accuracy that is determined by the direct comparison between the calculation results and the well-scaled experimental and plant data. The direct data comparisons produce the statement of code accuracy, i.e., mean deviation and the variance, which is then converted to important code uncertainty parameters and their statistics.

A variety of uncertainty analysis methods available for the parametric uncertainty analysis of deterministic computer models. Examples of such methods include response surface replacement, modified Monte Carlo, differential analysis, and distribution-free percentile estimation (DFPE). In this study the DFPE technique [8,9,10,11] with simple random sampling calculation (DFPE/SRSC) is used.

Uncertainty distributions of the results of a deterministic computer code results from the combination and propagation of the uncertainties associated with the code models and input parameters. They can be estimated by applying statistical methods in a logical and consistent manner. It is the aim of the DFPE/SRSC to obtain approximations to these distributions and derive quantitative uncertainty statements from them. To do this, a simple random sample is drawn from the selected code uncertainty parameters using their specified distributions.

An element of this sample is called the parameter vector and is composed of one value for each selected code parameters. The code is run with each parameter vector in the sample. The set of output values constitutes a simple random sample, which is drawn from the unknown probability distribution of the code calculation results. From this simple random sample, tolerance limit can be stated as a quantitative uncertainty measure. Tolerance limit [10] provide an interval within which at least a proportion q of the population lies, with probability $1\text{-}\alpha$ or more that the stated interval does indeed contain the proportion q of the population. A typical application of tolerance limits would be in a situation as follows:

A person is about to draw a random sample size of n, and he wants to know how large n should be so that he can be 95 % sure that at least 95 % of the population lies between the largest and smallest observations in the sample. But one-sided (upper-side) tolerance limits are of the following form. At least a proportion q of the population is greater than the largest observation with the probability $1\text{-}\alpha$. This tolerance limits can be directly applied to the present parameter uncertainty analysis. The minimum sample size n is determined by the following inequality:

$$1 - (0.95)^n \geq 0.95$$

$(0.95)^n$ is the probability for all n calculation output values not to exceed the true 95 percentile, $1-(0.95)^n$ is the probability for at least one of the calculation output values to exceed the true 95 percentile. Should there be some output values in the sample which do exceed the true 95 percentile, then the maximum output value is definitely one of those. Therefore, if the sample size meets the above inequality, then, one can be at least 95% confident that the true 95 percentile does not exceed the maximum of the *n* output values in the sample. The value of *n* that satisfies the above inequality is 59. It means that 59 code calculations for 59 randomly sampled uncertainty parameter vectors are necessary to conservatively estimate the true 95 percentile of the population of the code calculation results. This is referred as Wilks formula [6].

II-2. Steam Line Break Accident Scenario

In this study the CABUE methodology was applied to the SLB with offsite power available, in which the major safety concern is the reactivity margin after reactor trip. This is called as post-trip return-to-power SLB case. Offsite power is assumed to be available, which guarantees RCP running throughout the transient. Brief descriptions of the transient thermal-hydraulic characteristics and accident scenario are as follows:

A break in the main steam system piping causes an increase in steam flow from both steam generators. The increased steam flow increases the rate of heat removal from the steam generators and thereby decreases primary system pressure and temperatures decrease. The reactor core power rapidly increases mainly due to negative moderator temperature feedback effect. However, the reactor will trip by the actuation of reactor protection system. Possible reactor trip signals involved in this scenario include High Containment pressure (HCP), low Departure from Nucleate Boiling Ratio (DNBR), Variable Over Power (VOP), Low Steam Generator Pressure (LSGP), Low Pressurizer Pressure (LPP).

As blow-down of steam through the rupture continues, the steam generator pressure falls below the Main Steam Isolation Signal (MSIS) actuation pressure and Main Steam Isolation Valves (MSIVs) will be closed. The MSIV closure will stop the uncontrolled blow down from at least one steam generator. Auxiliary Feed Water (AFW) pumps start on the steam generator low level signal. Continued cool-down and associated shrinkage of reactor coolant causes the pressurizer pressure to drop below Safety Injection Actuation Signal (SIAS) and High Safety Injection Pumps (HPSIs) will deliver highly borated water to the Reactor Coolant System (RCS). If there is a return to criticality, core power peaking may occur, and the possibility of approaching to and exceeding the Specified Acceptable Fuel Design Limit (SAFDL) is the major safety concern.

II-3. Computer Code for Accident Analysis

RETRAN-3D/MOD3.1 [12] was used in the simulation of the accident. The RETRAN-3D code developed by EPRI is a best estimate transient thermal hydraulic code designed to analyze operational transients, small break LOCA, ATWS, natural circulation, long-term transients, and events involving limited non-equilibrium conditions in Light Water Reactors (LWRs).

The most important feature of RETRAN-3D code, that is very useful for uncertainty analysis, is the Steady State Initialization (SSI) capability. In some cases, uncertainty parameters affect the initial steady state condition and re-initialization for a steady state is needed when uncertainty parameters change. RETRAN-3D provides an automatic initialization feature to setup a steady state condition and it saves a lot of efforts and time needed to run a number of simulations in the uncertainty analysis like this study. In this study, separate 59 computer simulations were performed for a single scenario. The basic analysis model and specific break model for SLB are shown in Figures 1 and 2, respectively.

II-4. Identification of Uncertainty

A computer model [13] is an equation or a set of equations that represents a particular physical phenomenon within a computer program. This is product of various steps of human activities. Those steps are problem definition and conceptualization, formulation of the conceptual model, estimation of parameter values, and production of results. Every human activity is not free from mistakes, i.e. uncertainties. Therefore, each of the above five steps to develop and use a computer model inevitably has sources of uncertainties. Examples of considered uncertainty are critical flow model implemented in a code and operational uncertainty such as initial conditions of a plant.

At first such parameters that may include an uncertainty should be identified. Nine parameters were selected out of major influential parameters identified through an extensive PIRT process, and are presented in Table 1. The PIRT process performed for the APR-1400 was applied in this analysis. Since the system configuration of the APR-1400 is almost the same as UCN 3&4, the trends of overall plant responses to an accident in both plants are almost the same. Therefore, the PIRT results for the APR-1400 can be applied to the present analysis. Core physics data such as moderator density feedback, Doppler feedback, and scram rod worth and safety injection flow data are biased in a conservative manner due to lack of detailed information on the distribution types and the basis of data production.

The uncertainty distribution types and ranges of the identified input parameters should be specified. Uniform distribution is assumed for all parameters except the critical flow CD factor and critical flow model option. Critical flow model was selected from the two models implemented in the RETRAN-3D code. The ranges of the uniform distributed parameters are estimated mainly based on the design data generated for licensing calculation. Therefore, unavoidable conservatism is involved in the determination of distribution types and ranges of the uncertain parameters. A lot of study is yet to be done to prepare more reliable statistical data.

The covering check technique, which is a unique feature of CABUE, plays a key role in determining the ranges and distributions of the uncertainty parameter. This is an iterative procedure consisting of four steps. Thorough application of this procedure was not made in this paper due to insufficient SLB-related experimental data. A study to apply covering check technique is required to establish more reliable and licensable best estimate analysis methodology.

II-5. Analysis Results

Using the RETRAN-3D code , thermal hydraulic responses to the SLB was simulated for the 59 parameter vectors constructed by DFPE/SRSC. Also, a base case was run for nominal input data set (Table 1). The results were compared with the licensing calculation simulated by CESEC-III [14], a non-LOCA thermal hydraulic simulation code currently used for the KSNP type plants.

The calculation results are shown in Figures 3 through 8. The envelop of these calculations outlined by random sampling calculations can be argued, with 95% confidence, as the output parameters that are greater than the 95 percentile of the population.

Time variations of reactivity-related output parameters are shown in Figures 3 through 6. Total reactivity, which is the safety parameter in this scenario, is the sum of moderator and fuel temperature feedback reactivities, boron reacitivity, and scram reactivity. The figures indicate that the uncertainty propagates as the accident progresses by the interaction of various uncertainty parameters and computer model. Major contributors to the total reactivity are the moderator temperature feedback reactivity and the boron reactivity. The contribution of fuel temperature feedback reactivity is negligible. As expected, it can be shown that the base case calculation results fall within the envelope formed by random sampling calculation. The highest value of total reactivity out of random sampling cases is less than that of CESEC-III. This means that more reactivity margin to criticality after reactor trip can be obtained by applying the best estimate uncertainty analysis. Compared to the results of methodology, significant conservatism in the CESEC-III calculation with regard to the moderator temperature feedback and boron reactivities as shown in the figures. However, the fuel temperature feedback reactivity predicted by CESEC-III is smaller than that of random sampling cases. Detailed model comparison is needed to identify the reason for this difference.

Pressurizer pressure and core average temperature variations are shown in Figures 7 and 8. It can be seen that the uncertainty range of pressurizer pressure grows as the accident progresses, and that the responses of pressurizer pressure as predicted by RETRAN-3D and CESEC-III are quite different from each other. Since the pressurizer pressure determines the amount of boron reactivity insertion through safety injection and it has significant impact on the total reactivity, validity of the RETRAN-3D result needed to be confirmed through more study.

III. DISCUSSIONS

CABUE, a best estimate uncertainty quantification method, was applied to non-LOCA transient. It was shown that the combination of RETRAN-3D, a best estimate code and CABUE has the potential to enhance the safety margin compared with the existing conservative deterministic licensing analysis method. Even though the whole procedure of CABUE depends heavily on the computation, the current personal computers with giga-byte memory and with central processing unit speed of giga-instructions per seconds provide sufficient performance for the analysis.

As a first step in the application of best estimate uncertainty analysis method to non-LOCA events, the feasibility and usefulness of the method have been reviewed in this study. The success of this methodology relies on the adequacy of statistical data and identification process associated with the uncertainty parameters. In order to establish more reliable best estimate analysis methodology for non-LOCA transient analysis, more intensive and careful study is necessary on the identification of uncertainty parameters, determination of distribution types and ranges of the parameters, and computer model validation by comparison with plant transient data.

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Table 1. Uncertainty Parameters

Parameter	Distribution Type	Range	Base Case Value
Critical flow CD factor	Normal	±0.238	Mean
Critical flow option	DPD[1)]	1 of 2 [2)]	Option B
AFW flow rate	Uniform	500 – 800 gpm	650 gpm
MSIS setpoint	Uniform	851 – 975 psia	885.5 psia
Initial PZR pressure	Uniform	2000 – 2325 psia	2250 psia
Initial SG inventory	Uniform	35- 98.2 % WR	NWL[3)]
Inverse boron worth	Uniform	79 – 97 ppm/%Δρ	88 ppm/%Δρ
Safety injection delay time	Uniform	20 – 30 sec	25 sec
Initial PZR Liquid Volume	Uniform	21.9 – 60 %	52.6 %

1) Discrete Probability Distribution

2) Option A : Extended Henry (subcooled) and Moody (satuarated)

Option B : Isentropic Expansion HEM model

For detailed information, refer to [3]

3) Normal Water Level (79 %WR)

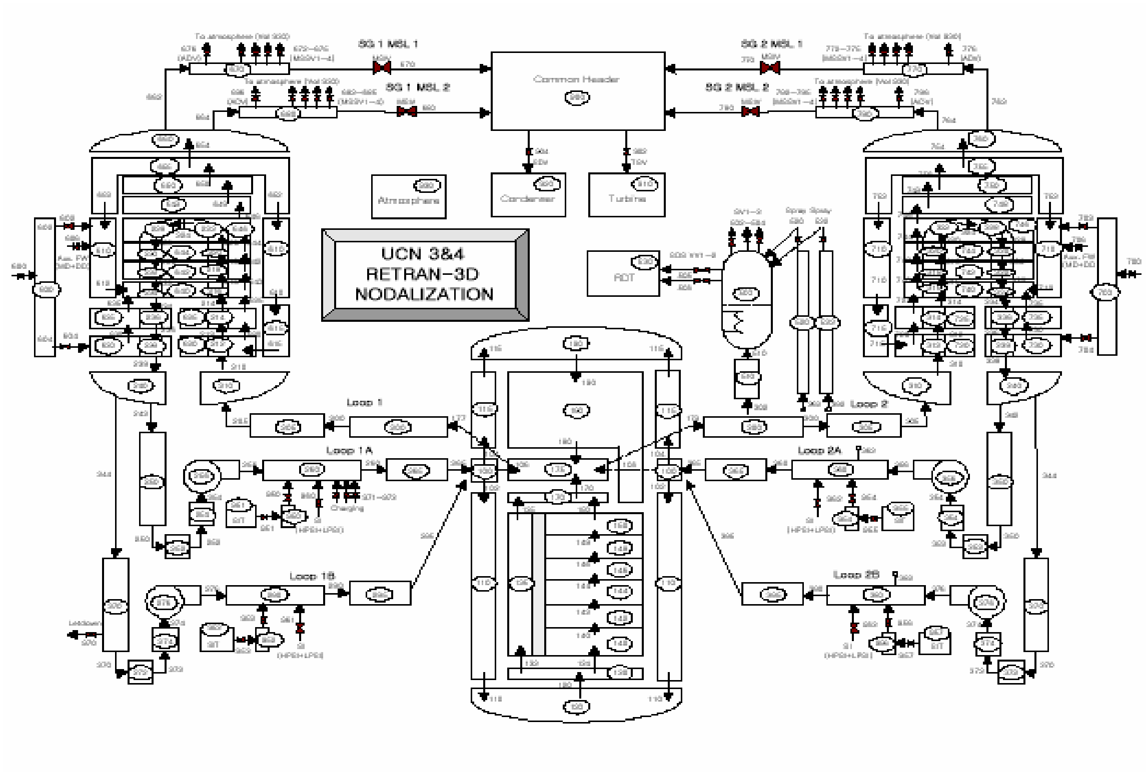

Figure 1. Basic RETRAN-3D Analysis Model for Steam Line Break

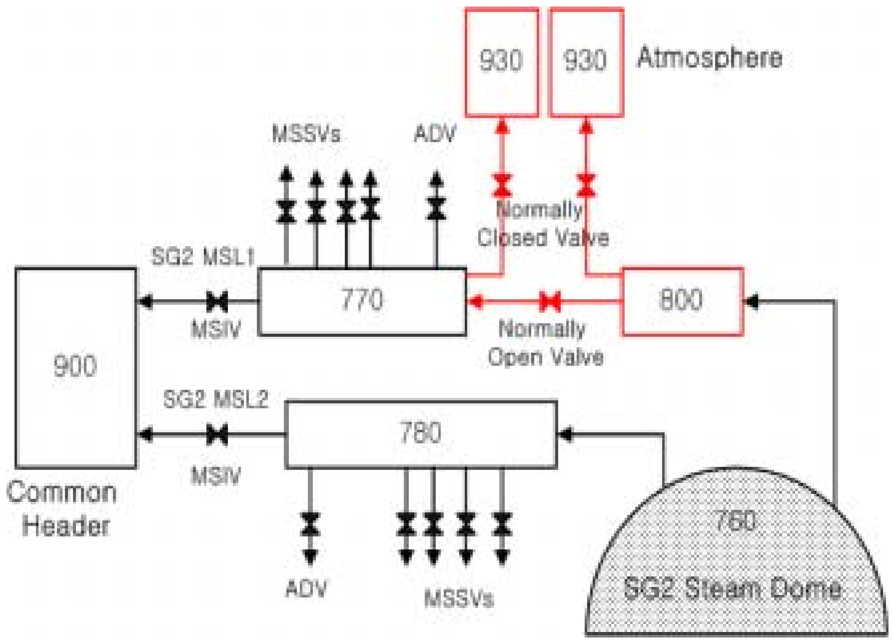

Figure 2. Break Model for Specific Steam Line Break Accident

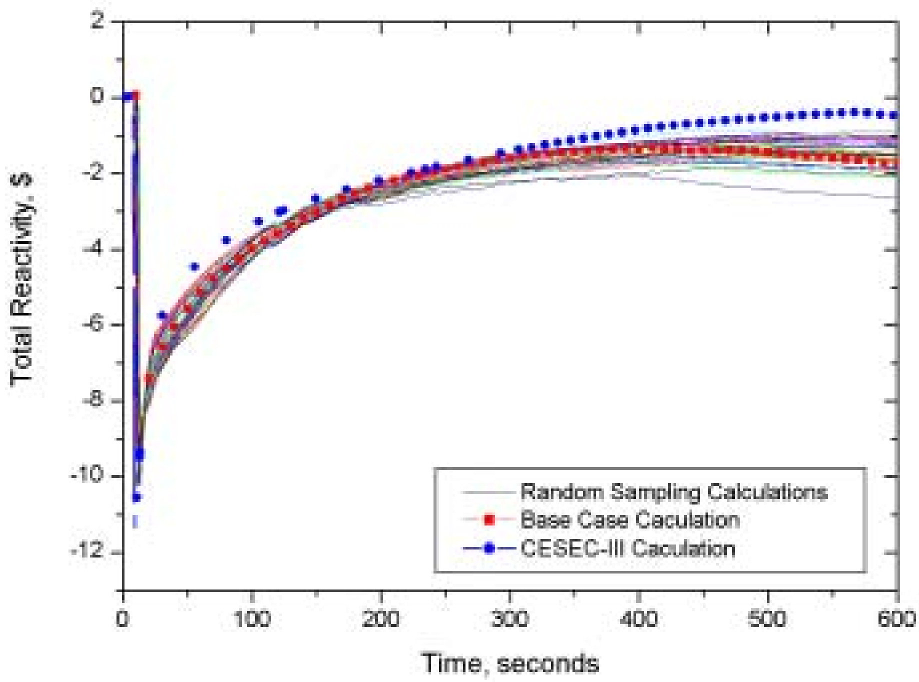

Figure 3.Total Reactivity Variation

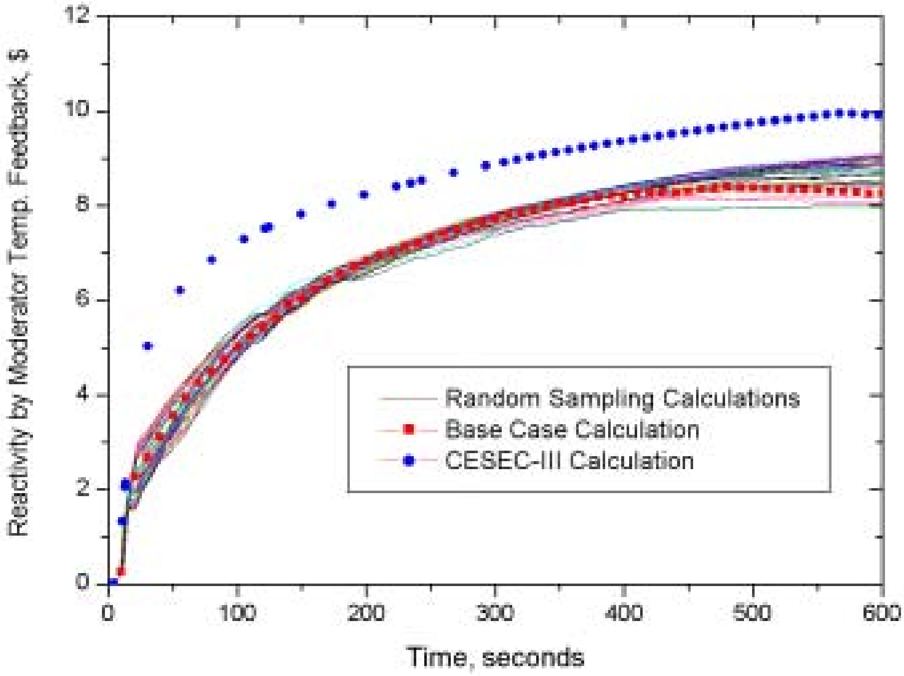

Figure 4. Moderator Temperature Feedback Reactivity Variation

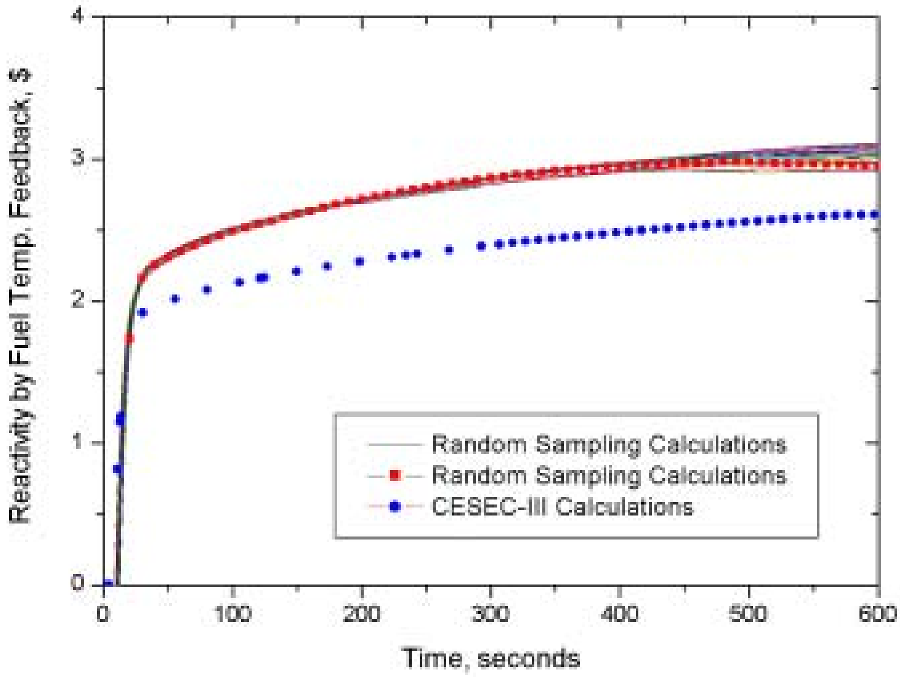

Figure 5. Fuel Temperature Feedback Reactivity Variation

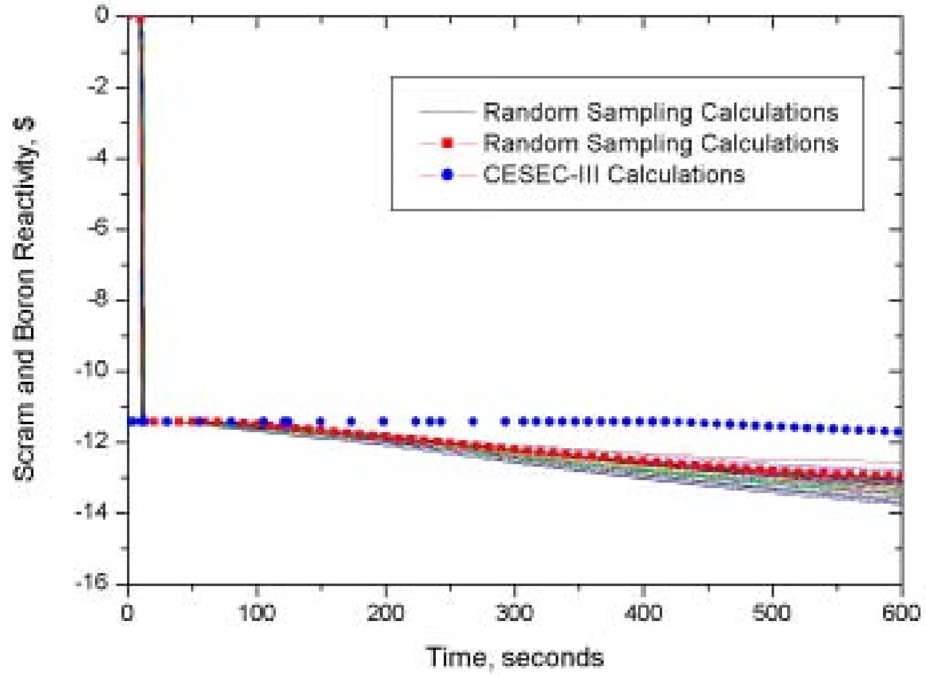

Figure 6. Scram and Boron Reactivity Variation

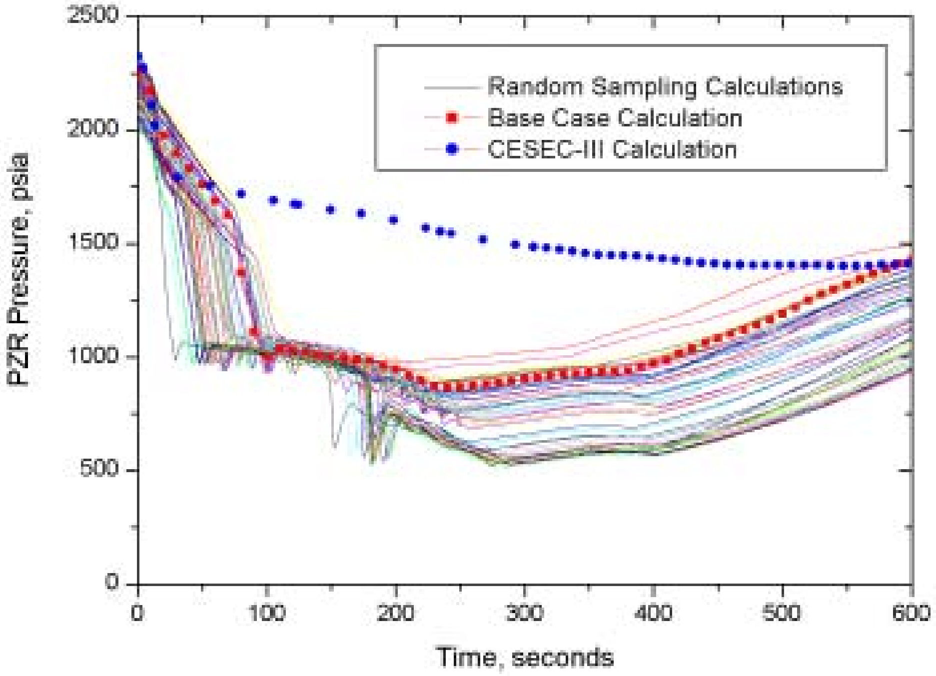

Figure 7. Pressurizer Pressure Variation

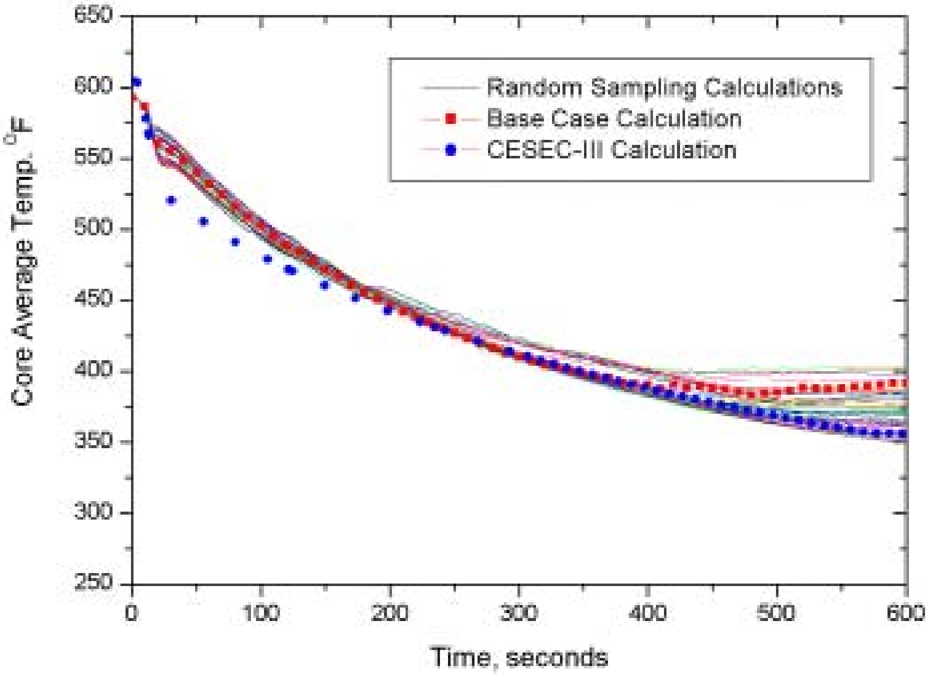

Figure 8. Core Average Temperature Variation