INL/CON-16-37561
PREPRINT

Limiting Rods Selection in Large Break LOCA Analysis with the RELAP5-3D Code

International Topical Meeting on Advances in Thermal Hydraulics (ATH '16)

Hongbin Zhang, Paul Bayless, Ling Zou, Haihua Zhao, Ronaldo Szilard

January 2016

LIMITING RODS SELECTION IN LARGE BREAK LOCA ANALYSIS WITH THE RELAP5-3D CODE

Hongbin Zhang, Paul Bayless, Ling Zou, Haihua Zhao and Ronaldo Szilard
Idaho National Laboratory
Idaho National Laboratory, P.O. Box 1560, Idaho Falls, ID, 83415-3870, USA
Hongbin.Zhang@inl.gov; Paul.Bayless@inl.gov; Ling.Zou@inl.gov; Haihua.Zhao@inl.gov; Ronaldo.Szilard@inl.gov

ABSTRACT

The Nuclear Regulatory Commission (NRC) is proposing a new rulemaking on emergency core system/loss-of-coolant accident (LOCA) performance analysis. In the proposed rulemaking, designated as 10 CFR 50.46(c), the US NRC put forward an equivalent cladding oxidation criterion as a function of cladding pre-transient hydrogen content. The proposed rulemaking imposes more restrictive and burnup-dependent cladding embrittlement criteria; consequently more fuel rods need to be analyzed under LOCA conditions to maintain the safety margin, compared to the current practice in which only one hot rod needs to be analyzed. New analysis methods are required to provide a thorough characterization of the reactor core in order to identify the locations of the limiting rods under LOCA conditions. In this work, the best-estimate plus uncertainty analysis capability for large break LOCA with the new cladding embrittlement criteria using the RELAP5-3D code is established and demonstrated with a reduced set of uncertainty parameters. With the new analysis method, the limiting transient case and the limiting rods can be easily identified in response to the proposed new rulemaking.

KEYWORDS
RELAP5-3D, 10CFR50.46(c), LBLOCA, BEPU, Limiting Rods

1. INTRODUCTION

The current emergency core cooling system (ECCS) acceptance criteria for loss-of-coolant accidents (LOCAs) in light-water reactors (LWRs) are described in 10 CFR 50.46. Two of the five criteria specify that the calculated peak cladding temperature (PCT) and maximum cladding oxidation shall not exceed 2200°F (1478K) and 17% equivalent cladding reacted (ECR), respectively [1]. Ever since the establishment of these cladding embrittlement criteria, more extensive research and experiments have been conducted which resulted in an increased understanding of fuel and clad behavior under both normal operating conditions and LOCA transient conditions. The new studies indicated that current regulatory acceptance criteria may be non-conservative for high burnup fuel. The Nuclear Regulatory Commission (NRC) is considering a rulemaking change that would revise the requirements in 10 CFR 50.46. In the proposed new rulemaking, designated as 10 CFR 50.46(c), the NRC proposed a fuel performance-based ECR criterion as a function of cladding hydrogen content before the accident (pre-transient), to include the effects of burnup on cladding performance [2]. The pre-transient cladding hydrogen content, in turn, is a function of the fuel burnup and cladding materials. A characteristic of the proposed new rulemaking, as illustrated in Figure 1, imposes more restrictive and fuel rod-dependent cladding embrittlement criteria; consequently fuel cladding performance and ECCS performance need to be considered in a stronger coupled way in LOCA analyses. Therefore, a thorough characterization of the reactor core is required in large break LOCA (LBLOCA) analyses. We aim at establishing such an analysis capability with the RELAP5-3D code [3].

In this study, we will present the methods and results from the demonstration of applying the best-estimate plus uncertainty (BEPU) methodology using the RELAP5-3D code to a LBLOCA analysis. The BEPU methodology has become the de-facto industry standard for LOCA analyses, which relies on using the Wilks' nonparametric statistical approach to demonstrate compliance with safety criteria. Wilks' non-parametric statistical approach randomly samples all the uncertainty parameters in the space defined by the uncertainty ranges and combines all the uncertainties such that the combined effect of the uncertainty parameters to the total uncertainty can be quantified in the LOCA analysis. It uses a relatively small number of samples of LOCA transient calculations to obtain the limiting transient case as well as the 95/95 (95% probability with 95% confidence level) upper tolerance limits of the safety metrics.

In the current practice of LOCA analysis with 10CFR 50.46 rules, the limiting fuel rod in the limiting transient case can be easily identified with the safety metric being the fuel rod with the highest PCT, which is normally the highest power rod (hot rod). The reactor core modeling in a LBLOCA analysis normally uses a simplified approach with the core flow represented by a hot channel and an average channel. The hot channel represents the flow channel adjacent to the highest power rod and the average flow channel represents the remaining flow in the core. The fuel and clad temperature distributions and clad oxidation rates within the hot rod are calculated by building a heat structure for the rod and attaching it to the hot channel. Average heat structures are built for the remaining fuel rods in the core and attached to the average flow channel such that the fuel and clad temperature distributions and clad oxidation rates can be calculated for the average rods. Conversely, with the proposed new rulemaking in 10 CFR 50.46(c), both the PCT and ECR safety acceptance criteria are functions of the pre-transient hydrogen content in the clad. The limiting rods may not be the hot rod (highest power rod) anymore and could even move from one rod location to another depending on fuel burnup and other conditions in an operating cycle. Therefore, new safety metrics have to be defined in compliance with 10 CFR 50.46(c) and all the fuel rods have to be considered in LOCA analyses in order to identify the limiting rods. In this work, a systematic approach to search for the limiting rods is developed in the context of BEPU analysis of LBLOCA using the RELAP5-3D code.

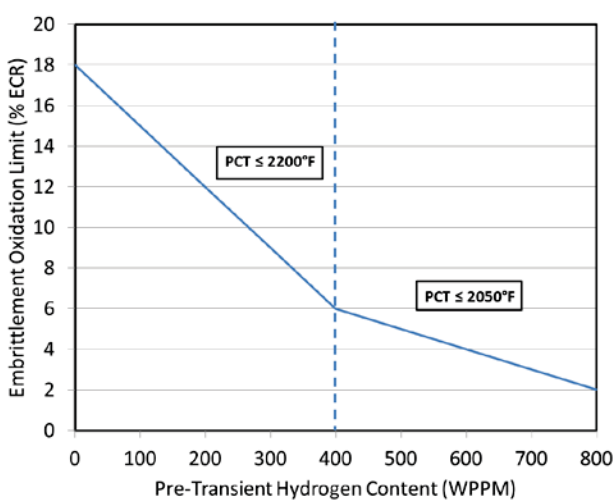

Figure 1. Analytical Generic Limit Proposed by the NRC for Existing Fuel, ECR & PCT versus Hydrogen [2].

2. SYSTEMATIC METHOD TO SEARCH LIMITING RODS IN LBLOCA ANALYSIS WITH RELAP5-3D

A typical four-loop pressurized water reactor (PWR) with 3411 MW rated thermal power has been selected for analysis with RELAP5-3D. The accident scenario selected is a LBLOCA with a double-ended guillotine break in a cold leg. The nodalization diagram of the RELAP5-3D model for the typical PWR is shown in Figure 2. The safety criteria are the generic acceptance criteria for the peak clad temperature and the maximum oxidation rate (as shown in Figure 1) proposed in the new rulemaking. Since both PCT and ECR limits are burnup-dependent, this added complexity requires defining new safety metrics that would synthesize PCT and ECR with fuel rod dependent cladding pre-transient hydrogen content. In this work, the safety metrics are defined as the ratios of the calculated PCT over PCT limits for each fuel rod, as well as the ratios of the calculated ECR over ECR limits for each fuel rod and are expressed in the following:

$$PCTR = \frac{PCT^{Calculated}}{PCT^{Limit}} \tag{1}$$

$$ECRR = \frac{ECR^{Calculated}}{ECR^{Limit}} \tag{2}$$

If we define $PCTR_{max}$ and $ECRR_{max}$ as the maximum value of PCTR and the maximum value of ECRR, respectively, the acceptance criteria for the safety metrics are the following:

1) $PCTR_{max} < 1.0$

or

2) $ECRR_{max} < 1.0$

Using the above criteria, the limiting fuel rods can be identified as the fuel rods with $PCTR_{max}$ or the $ECRR_{max}$.

The reactor core has 193 assemblies with three batches of fuel loading – the fresh fuel, once-burnt fuel and twice-burnt fuel. An equilibrium cycle was achieved for this core design [4] and the results were fed into the RELAP5-3D model. The details of the core physics analysis tools and core design parameters can be found in [4].

The reactor core modeling in RELAP5-3D used different homogenization approaches for thermal fluid dynamics calculations than for the heat conduction and clad oxidation calculations in the fuel rods. A multiple channel approach was used for the thermal fluid dynamics calculation, as illustrated in Figure 3. Specifically, the assemblies in the core were grouped into various regions based on their burnup history. The assemblies with fresh fuel, once-burnt fuel and twice-burnt fuel were grouped together respectively. Two flow channels – one average channel and one hot channel – were built to represent each group of assemblies. Hence there are a total of six flow channels in this study. The flow channels are connected in the lateral direction to allow crossflow to be calculated. Crossflow is modeled at each axial elevation in the core between the three average core channels. It is also modeled at each axial elevation between the hot channels and the adjacent average channels. This allows flow to be redistributed around a blockage caused by cladding ballooning or rupture. The crossflow area is based on the minimum gap between the fuel rods along one side of a fuel assembly and the number of fuel assembly sides at the interface between the three average core channels; for example, for the hot assembly in each region, there are four sides at the interface. Loss coefficients are approximated based on flow across in-line and staggered rows of tubes, with the average distance of travel estimated to be about half an assembly width.

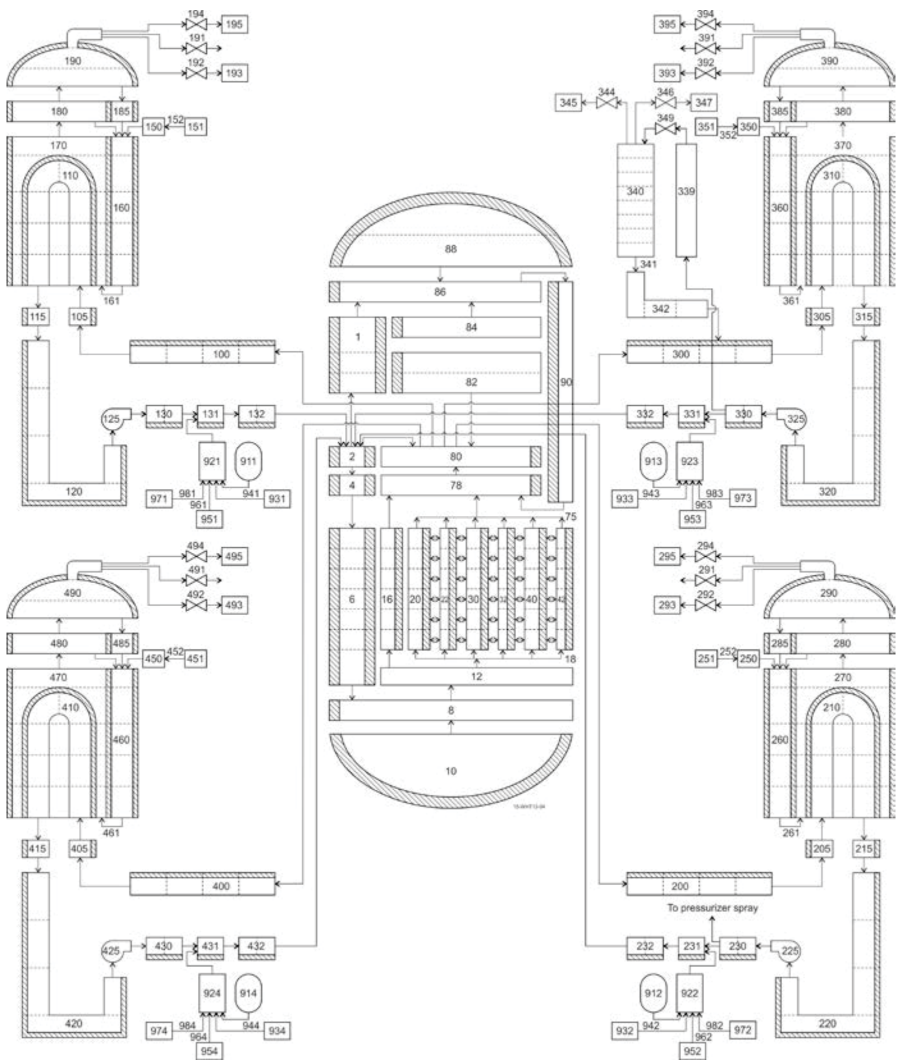

Figure 2. RELAP5-3D Nodalization of a Typical Four-Loop PWR

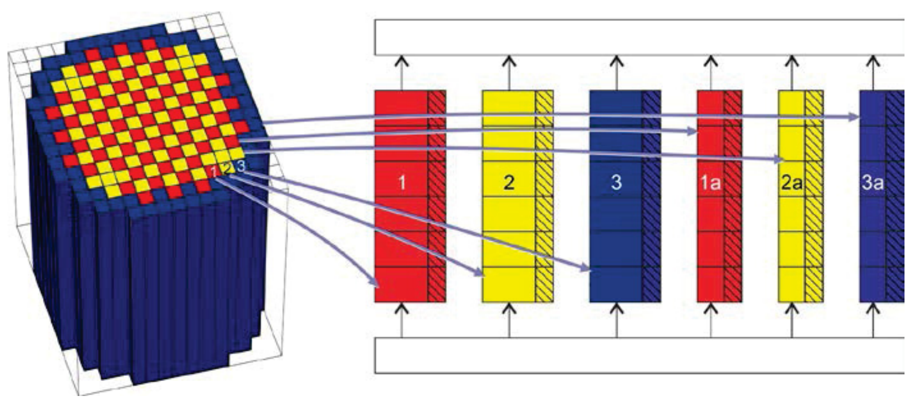

Figure 3. Schematic Illustration of the Mapping Between the Core Design Analysis and the RELAP5-3D Analysis Core Model for a Typical Four-Loop PWR

For heat conduction and clad oxidation calculations, it is computationally prohibitive to consider all the fuel rods in the reactor core. Instead a homogenization technique is used to reduce the number of fuel rods to be simulated. Two sets of heat structures were used for each assembly – one set represents the highest power rod or the hot rod in the assembly and the other set represents the average of the remaining fuel rods in the assembly. This is a reasonable approximation given that the fuel rod burnup normally does not vary too much within a PWR assembly and the hot rod in an assembly would be the limiting rod for that assembly.

As a result, heat structures for the highest power assembly (hot assembly) and its hot rod in each group of assemblies were built and attached to the hot channel, as shown schematically in Figure 4, such that the PCT and ECR in the average rods and hot rod can be calculated. Analogously, the heat structures for the other assemblies and their respective hot rods were built and connected to the average channel, as shown in Figure 5, such that the PCT and ECR can be calculated for the average rods and hot rod in each assembly. Therefore, there are a total of 386 sets of heat structures for the fuel in this study (193 for assemblies plus 193 for hot rods). It is noted that the hot rod power has been subtracted from each assembly to yield the correct power for the average fuel rods in each assembly such that the reactor total power is conserved.

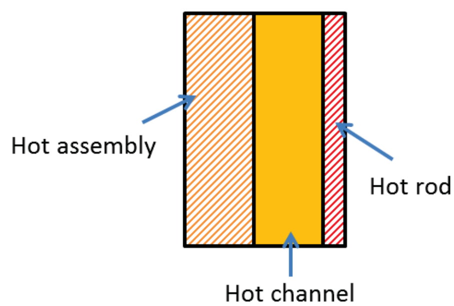

Figure 4. Schematic Illustration of the Heat Structure Mapping for the Hot Assembly and Its Hot Rod With the Hot Channel (One for Each Group of Assemblies)

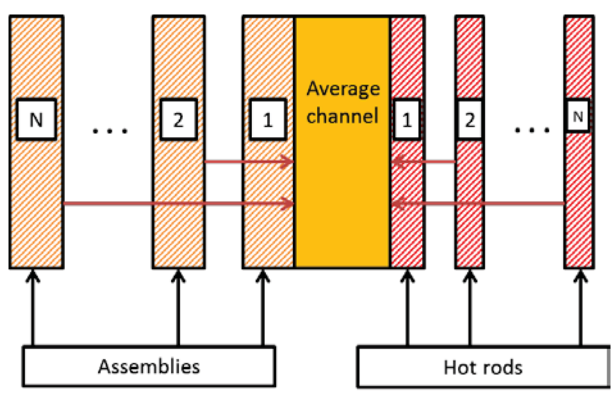

Figure 5. Schematic Illustration of the Heat Structure Mapping for Average Assemblies and Their Respective Hot Rods with the Average Flow Channel

In order to perform BEPU analysis, the important phenomena affecting the progression of the LBLOCA accident are first determined by the phenomena identification and ranking table (PIRT) process. A large number of studies have been done previously to identify the important phenomena. A PIRT analysis has been conducted in this work with input from FPoliSolutions LLC [4]. For demonstration purposes, a reduced set of parameters with high importance to LBLOCA has been selected and is shown in Table I.

Table I. Distribution of parameter uncertainties

Parameter	PDF type	Min	Max	Comments
Reactor thermal power	Normal	0.98	1.02	Multiplier
Reactor decay heat power multiplier	Normal	0.94	1.06	Multiplier
Accumulator pressure	Normal	-0.9	1.1	Multiplier
Accumulator liquid volume (m^3)	Uniform	-0.23	0.23	Additive
Accumulator temperature (K)	Uniform	-11.1	16.7	Additive
Subcooled multiplier for critical flow	Uniform	0.8	1.2	Multiplier
Two-phase multiplier for critical flow	Uniform	0.8	1.2	Multiplier
Superheated vapor multiplier for critical flow	Uniform	0.8	1.2	Multiplier
Fuel thermal conductivity	Normal	0.93	1.07	Multiplier
Average core coolant temperature (K)	Normal	-3.3	3.3	Additive
Film boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier

The Wilks' non-parametric statistical sampling technique [5] is used to determine the 95/95 upper tolerance limits from unknown distributions by randomly sampling the uncertain input parameters shown in Table I. If only one outcome, such as PCT, is considered, using the Wilks' formula, the sample size can be determined for a desired population proportion at a given tolerance interval by:

$$\beta = 1 - \gamma^N \tag{3}$$

where β is the confidence level, γ is percentile and N is the number of RELAP5-3D runs required. For instance, if we are interested in determining a 95/95 bounding value of PCT (γ=0.95) with 95% confidence level (β=0.95), N is found to be 59 from Wilks' formula. In this technique, all the uncertainty parameters are sampled simultaneously in each RELAP5-3D run. The PCTR results are then ranked from the lowest to the highest such that the limiting case can be identified. The 95/95 value can be conservatively estimated by the corresponding PCT value of the largest PCTR value from 59 runs. A more general formula than Equation 3 allows a less conservative limiting value, such as the 2nd largest value of PCTR from 93 runs, the 3rd largest value of PCTR from 124 runs, etc. Since the new ECCS/LOCA rulemaking is also more restrictive on the local ECR, at a minimum the outcomes of PCT and ECR need to be considered jointly to provide a probability statement of compliance with respect to the acceptance criteria. According to Frepoli [6], the sample size needs to be extended beyond 59 and can be found by solving the following equation for N:

$$\beta = \sum_{j=0}^{N-p} \frac{N!}{(N-j)!j!} \gamma^j (1-\gamma)^{N-j} \tag{4}$$

where p is the number of outcomes considered. For our analysis, the safety metrics are the maximum PCTR and ECRR. According to Ref. [6], by substituting γ=0.95, β=0.95 and p=2 into the above equation, N is found equal to 93. The limiting cases for the 93 RELAP5-3D runs are the cases with the maximum PCTR value or ECRR value.

Since a LOCA event is equal-probable in time, the time in a cycle is an additional random variable whose uncertainty is propagated through the analysis in a typical LBLOCA analysis conducted in the current practice [4]. Including the time in a cycle as a random variable would require LOCA calculations to be carried out on a very large number of exposure points, which is not practical. As a result, in our demonstration calculations we selected a few specific exposure points in a cycle and then propagated all the uncertainties for each exposure point.

The exposure points selected for the LBLOCA calculations cover the entire range of the cycle length. The selected exposure points are at the beginning of cycle (BOC), 100 days, 200 days, 300 days, 400 days, 500 days and end of cycle (EOC). This way, the dynamic response of the plant with regards to LBLOCA transients would be fully characterized at different core conditions during the entire cycle.

A set of 93 RELAP5-3D input files has been prepared respectively at each of the seven selected exposure points by randomly perturbing the input parameters using their associated probability density functions defined in Table I. All the RELAP5-3D cases were run to steady state first. Large break LOCA cases were then initiated by assuming a double-ended guillotine break. Following the initiation of the LBLOCA, a fast depressurization of the primary system ensues. The ECCS is activated to provide emergency cooling water to the core. The entire process lasts about 10 minutes. To be conservative, in our RELAP5-3D plant model simulations, the shutdown of the reactor following the initiation of LBLOCA is achieved through the negative reactivity feedback, rather than through the scram of the reactor. In our LBLOCA runs, it is assumed that only two out of the three ECCS systems in the intact loops are functioning and able to inject water into the reactor core. However, as passive components, it is assumed that all three accumulators in the intact loops are functioning and able to inject water into the reactor core.

RELAP5-3D contains simplified fuel performance models and uses the Cathcart-Pawel oxidation model [7] to do the clad oxidation calculations in the LOCA runs. The fuel rod deformation and rupture model is also turned on in the RELAP5-3D simulations such that the cladding inner surface oxidation can be accounted for when a fuel rod ruptures during the LOCA transients. The ECR is calculated by dividing the cumulative hydrogen generated in each axial segment of the fuel rod heat structure by the mass that would be generated if all of the cladding in that segment were oxidized.

A LOCA analysis toolkit has been developed using the Python programming language. The LOCA toolkit automatically samples each uncertain parameter shown in Table I from its distribution. For a uniform distribution, the minimum and maximum values are the boundaries of the sampling. For a normal distribution, the sampling boundaries were truncated at the minimum and maximum values, which is effectively a truncated normal distribution. No dependencies between parameters were considered in the sampling. The LOCA toolkit then modifies the RELAP5-3D input files according to the perturbed values. It automatically drives the desired number of RELAP5-3D runs on Idaho National Laboratory's high performance computers (HPC). The toolkit also performs the postprocessing of the RELAP5-3D output files and presents the $PCTR_{max}$ and $ECRR_{max}$ values according to Wilks' approach. Figure 6 illustrates the data flow process with the BEPU analysis toolkit.

3. LBLOCA ANALYSIS RESULTS

The maximum PCT and ECR values for each assembly and its hot rod in the core are retrieved by the toolkit from each of the 93 RELAP5-3D output files at each selected cycle exposure point. For each RELAP5-3D output file, the toolkit then calculated the PCTR and ECRR values for each assembly and its hot rod. To calculate the PCTR and ECRR, the cladding hydrogen content at pre-transient condition is needed. The hydrogen content is a function of fuel burnup and cladding materials. Since detailed fuel performance calculations have not been included yet in this work, the pre-transient hydrogen content is not readily available. Instead, for demonstration purposes, Figure 3 of Ref. [1] was used to correlate the pre-transient hydrogen pickup content (WPPM) with the fuel rod average burnup for Zircaloy-4 cladding. Figure 3 of Ref. [1] is reproduced and shown in Figure 7. A curve fitting for Figure 3 of [1] was done to yield the following:

$$H = 0.0035BU^3 - 0.1235BU^2 + 5.1073BU + 21.075 \quad (5)$$

where H is the hydrogen content in WPPM and BU is the fuel rod average burnup in GWD/MT. The core design tool that was used for this work does not have the capability yet to compute rod average burnup; instead we assumed the rod average burnup in an assembly is the same as the assembly average burnup. Equation (5) was used to correlate the pre-transient hydrogen content in one assembly as a function of assembly average burnup calculated from core design activities.

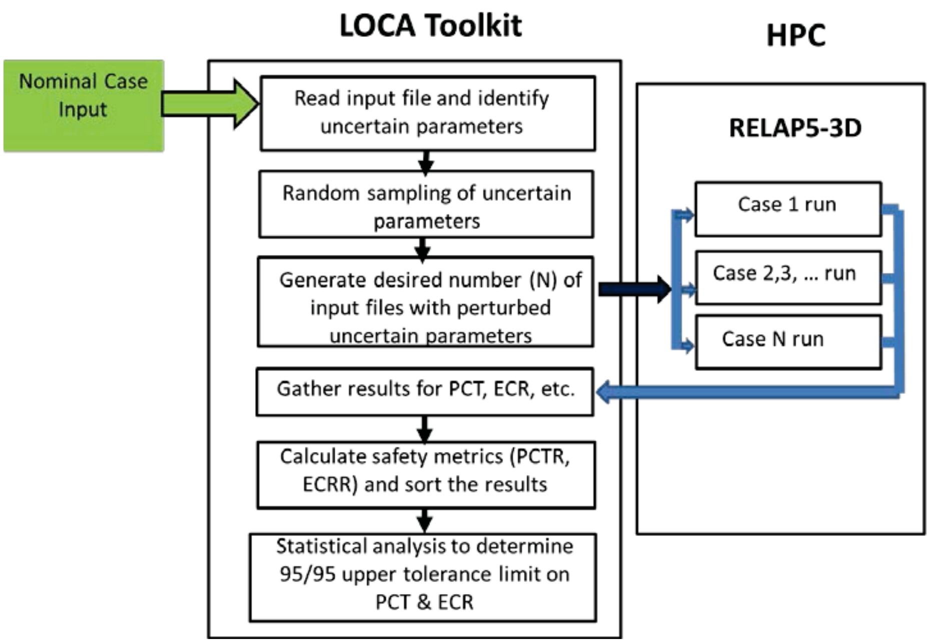

Figure 6. Diagram of Data Flow Process with the LOCA Toolkit

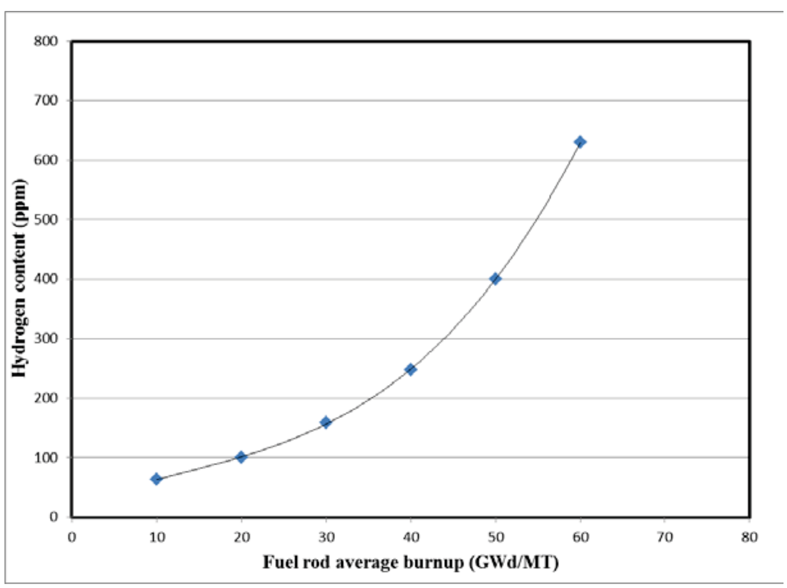

Figure 7. Zircaloy-4 Hydrogen Pickup versus Burnup (Reproduced from Figure 3 of [1]).

The LOCA toolkit calculated the pre-transient hydrogen content in the clad according to Equation (5). The WPPMs were subsequently used to find the limiting values of PCT and ECR (shown in Figure 1) for use in the PCTR and ECRR calculations. For each of the 93 RELAP5-3D LBLOCA calculations at one selected exposure point, the $PCTR_{max}$ and $ECRR_{max}$ values were obtained by sorting all the PCTR and ECRR values in the core. Subsequently, the $PCTR_{max}$ and $ECRR_{max}$ values obtained from each RELAP5-3D run were further sorted among the 93 runs. The cases with the maximum values of $PCTR_{max}$ or $ECRR_{max}$ among the 93 runs are considered to be the limiting cases and the fuel rods with the maximum values of $PCTR_{max}$ or $ECRR_{max}$ are the limiting fuel rods at the selected cycle exposure point.

To demonstrate compliance with the proposed new rulemaking for ECCS/LOCA, it is desirable to show the PCT and ECR values as a function of the pre-transient hydrogen content for the limiting cases identified in the LBLOCA runs. The PCTR and the PCT values versus the pre-transient cladding hydrogen content for all the assemblies in the core for the limiting cases at the seven selected cycle exposure points are shown in Figs. 8 and 9 respectively. The ECRR and ECR values versus the pre-transient cladding hydrogen content for all the assemblies in the core for the limiting cases at the seven selected cycle exposure points are shown in Figs. 10 and 11, respectively.

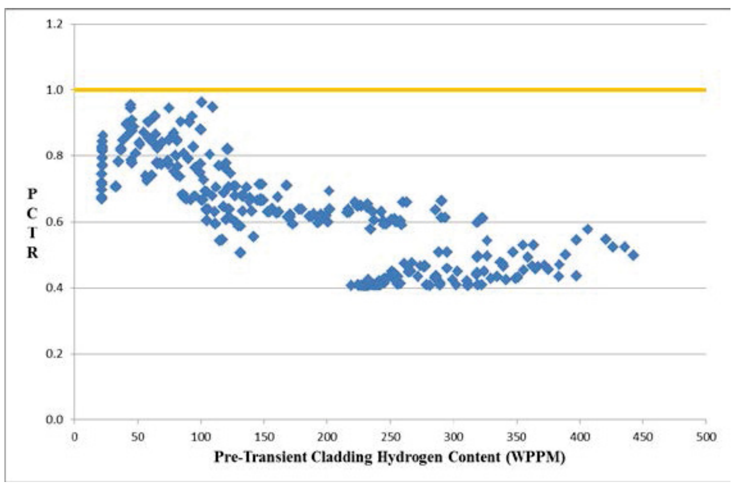

Figure 8. PCTR Distributions of the Limiting Cases at the Seven Selected Exposure Points.

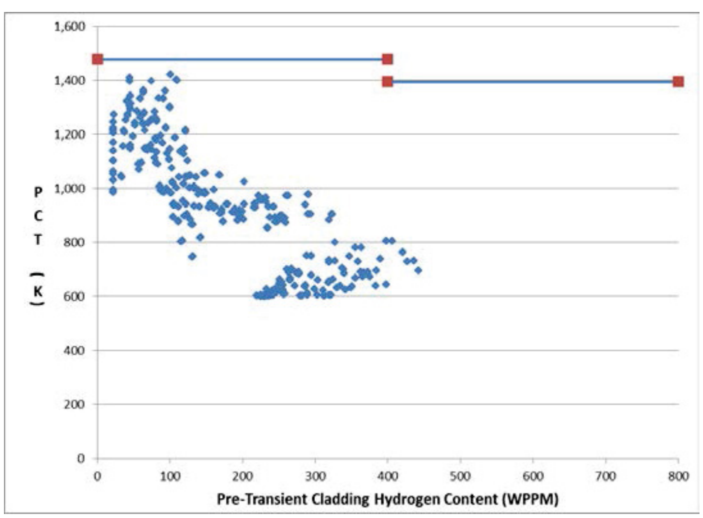

Figure 9. PCT Distributions of the Limiting Cases at the Seven Selected Exposure Points.

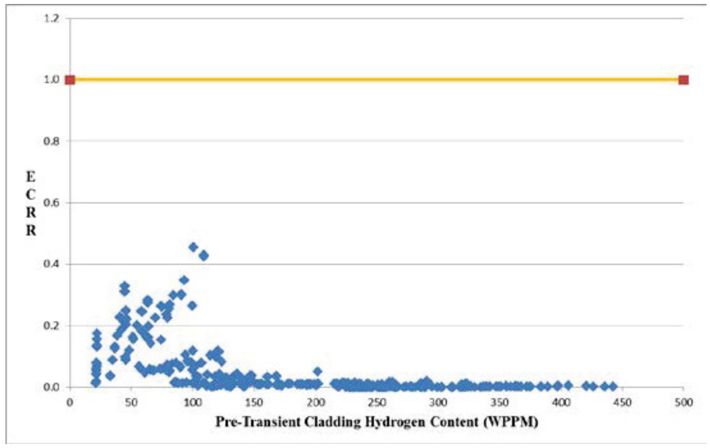

Figure 10. ECRR Distributions of the Limiting Cases at the Seven Selected Exposure Points.

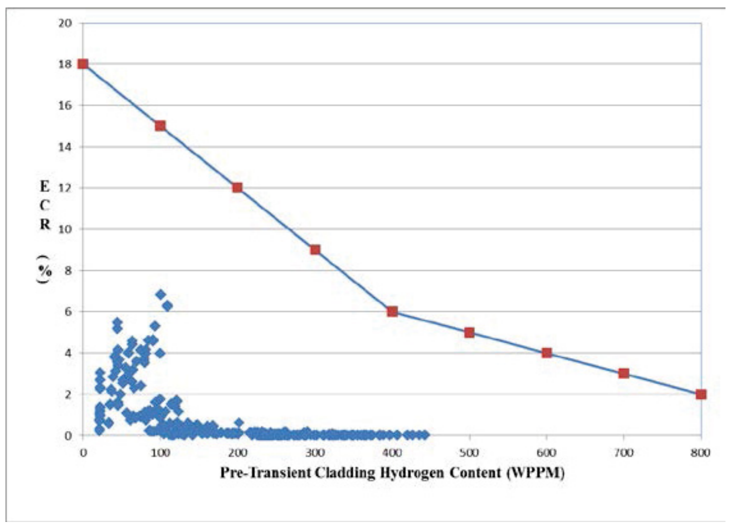

Figure 11. ECR Distributions of the Limiting Cases at the Seven Selected Exposure Points.

4. CONCLUSIONS

The BEPU methodology has been demonstrated using the RELAP5-3D code for a postulated large break LOCA transient for a typical four loop PWR. The ordered non-parametric statistical approach is used to determine the statistical upper limit for PCT and ECR. A methodology to identify the limiting fuel rods in response to NRC proposed 10 CFR 50.46c new rulemaking has been developed and demonstrated. Future work includes incorporating fuel performance calculations into the RELAP5-3D model to provide more realistic steady-state initialization of the fuel rod conditions. Other future work includes combining the probabilistic risk analysis and BEPU analysis to perform risk-informed LBLOCA analysis for the NRC's proposed new rulemaking on the ECCS/LOCA performance acceptance.

ACKNOWLEDGMENTS

This work is supported by the U.S. Department of Energy, under Department of Energy Idaho Operations Office Contract DE-AC07-05ID14517. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes. The authors are grateful to Cliff Davis at INL and Cesare Frepoli from FPoliSolutions LLC for their valuable input to improve the RELAP5-3D models. The authors wish to thank Angelo Zoino and Andrea Alfonsi for the core design work they have performed.

REFERENCES

1. http://pbadupws.nrc.gov/docs/ML1114/ML11140A159.pdf, Letter Report: OG-11-143 PWROG 50.46(b) Margin Assessment.
2. http://pbadupws.nrc.gov/docs/ML1111/ML111100391.pdf.
3. RELAP5-3D Code Development Team, 2015, *RELAP5-3D Code Manual, Volumes I-V*, INL/MIS-15-36723, Revision 4.3, October 2015.
4. R. Szilard, et al., *Industry Application Emergency Core Cooling System Cladding Acceptance Criteria Early Demonstration*, INL/EXT-15-36541, September 2015.
5. S. S. Wilks, "Determination of Sample Sizes for Setting Tolerance Limits," *The Annals of Mathematical Statistics*, Vol. 12, no. 1, pp. 91-96, 1941.
6. C. Frepoli and S. Iyengar, "A Comparison of Non-Parametric Tolerance Limits with Linear Combinations of Order Statistics in Safety Analysis," *Proceedings of ICAPP*, Nice, France, May 2-5, 2011.
7. J. V. Cathcart, et al., "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge, Doc. No. ORNL/NUREG-17, ADAMS Accession no. ML052230079, 1977.