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Quantitative Analysis of Safety Margin Reduction of Power Uprate in SBLOCA

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Abstract

Nuclear Energy Agency of OECD has developed a methodology in the Safety Margin Action Plan (SMAP) to quantify the reduction of safety margin due to power uprate. The methodology combines the probabilistic and deterministic approaches of safety analyses. In the present study, the SMAP methodology is adopted to quantify the reduction of safety margin of a 5% power uprate of Maanshan nuclear power plant (NPP) of Taiwan Power Company. The sequence selected to demonstrate the methodology is small break cold leg loss of coolant accident. An evaluation model code, RELAP5-3D/K is used in the LOCA analysis. The plant specific probabilistic safety assessment (PSA) model is used to determine the accident scenario analysed. Uncertainty of the predicted peak cladding temperature (PCT) during the accident is quantified using the rank statistics. Phenomenon Identification and Ranking Table (PIRT) for parameters that are important for the small break LOCA analyses and their distribution are identified. Two sets of calculations are performed to determine PCT of the 95th percentile with 95% of confidence at 102% and 105% power cases. The results demonstrate that, statistically, the safety margin for the particular accident scenario analysed is not reduced due to power uprate from 102% to 105% power.

Introduction

There are two ways to increase the potency of operation of nuclear power plant. The deduction of operation cost and the increase of power generated. Power uprate is one of the methods to increase the amount of power generation. We can expect that the safety margin of the plant will be reduced in the power uprate. Traditionally, the Licensing Evaluation Models as specified in the regulatory arena are used to assess the acceptability of power uprate There are all kinds of conservatism embedded in the safety criteria and the Evaluation model in the licensing calculations. The calculations can assure that the safety criteria are not violated after power uprate. Nevertheless, the deduction of safety margin due to power uprate can not be quantified in the calculations using Evaluation Model.

In the Safety Margin Action Plan [1] of OECD/NEA, a methodology has been developed to quantify the reduction of safety margin due to power uprate. The methodology combines the probabilistic and the deterministic approaches of safety analysis. In SMAP, safety margin is represented by uncertainty distributions of target parameter. Uncertainty can be separated into two parts: aleatory and epistemic. The aleatory uncertainty is the uncertainties of the performances of the system and its components. The aleatory uncertainty is treated probabilistically using the approach of Fault Tree and Event Tree analyses. Epistemic uncertainty is caused by the "imperfect knowledge" regarding the computation models and values of input parameters of these computational models. Therefore, the epistemic uncertainty also has two parts: computation models and value of input parameters.

In this paper, the SMAP methodology is adopted to quantify the reduction of safety margin of a 5% power uprate of Maanshan nuclear power plant of Taiwan Power Company. The plant employs a Westinghouse designed three-loop Pressurized Water Reactor (PWR). The original rated power of the plant is 2775 MWt. The event analyzed is a cold leg small break loss of coolant accident with a diameter of 3 in. The peak cladding temperature is selected as the target parameter of the safety margin reduction analysis. Thermal hydraulic calculation of the analysis is performed using RELAP5-3D/K code, which is an Evaluation Model code. The code was developed by Institute of Nuclear Energy Research of Taiwan based on MRELAP5-3D code. The plant specific PSA model of Maanshan NPP is adopted in the analyses. Realistic values of input parameters from Monte Carlo sampling, instead of the conservative values as required in the Appendix K of 10CFR 50, are used in the present analyses. The uncertainty due computation model is not quantified in this paper.

1. Thermal-Hydraulic Model

1.1 Thermal-Hydraulic Analysis Code

RELAP5-3D code for nuclear power plant thermal-hydraulic analysis is developed by Idaho National Lab (INL) under the support of U.S. Department of Energy (DoE). RELAP5-3D analysis program is mainly used in the transient and LOCA analyses of Light Water Reactor (LWR). RELAP5-3D considers the reactor coolant system as series or parallel control volumes connected by junctions. The control volumes can be modeled three dimensionally. RELAP5-3D provides special component models for nuclear power systems such as pump, valve, pipe, steam separator, turbine etc. RELAP5-3D also has the capability of modeling the control logic of the plant. RELAP5-3D/K is a revised version of RELAP5-3D. The requirements for the analyses of loss of coolant accident as specified in the Appendix K of 10CFR 50 are incorporated into the code by Institute of Nuclear Energy Research of Atomic Energy Council of Taiwan.

1.2 Nodalization

Figure 1 shows the RELAP5 nodalization diagram of Maanshan Nuclear Power Plant. The Input deck was developed by Nuclear System Kinetic Modeling and Analysis Laboratory of National Tsing Hua University. The input deck includes 246 interconnected control volumes, 272 flow junctions and 199 heat structures. The components modelled include the reactor pressure vessel, reactor coolant pumps and related pipings, the primary and secondary side of three steam generators (SG), power operated relief valves and safety valves of SG secondary side, main steam line isolation valves, steam dump system, pressurizer and its major component for pressure regulating, and accumulators. The turbine, injection of emergency core cooling system, and auxiliary feedwater system are modelled using time dependent volume or junction. Point Kinetic Model is selected to simulate the behaviour of neutrons in the transient.

1.3 Initial Condition and Power Uprate Condition

The original rated power of the plant analyzed is 2775 MWt. In 2009, the plant implemented the measurement uncertainty recapture power uprates (MUR). The rated power raised to 102%. Therefore, the base case in the present analysis is 102%. According to the current plant status, 10% of the U-tubes has been plugged.

Before the power uprate, the system rated power is 2830 MWt. Pressurizer pressure is 2250 psia. Thermal design flow rate is 92,600 gpm/loop. RCS flow rate is 28121 lbm/s, and core bypass rate is 6.8%. The hot leg is 624.9°F and the cold leg temperature is 556.4°F. Feedwater flow rate is 3559.3 lbm/s with a temperature of 440°F. The pressure of SG secondary side is 980 psia and the steam flow rate is 1186.7 lbm/s.

After stretch power uprate (SPU) to 105%, the rated power is 2914 MWt. It is assumed that pressurizer pressure, thermal design flow rate, RCS flow rate, core bypass rate, feedwater flow rate and its temperature, and the pressure of steam generator secondary side all keep at the same values. The cold leg temperature is 556.6°F and the hot leg temperature raise to 626.8°F to take away more heat. The steam flow rate increases to 1221.3 lbm/s.

The initial conditions before and after power uprate as predicted by the steady state initialization of RELAP5-3D/K code are listed in Table 1.

Parameter	102% MUR	105% SPU
Rated Power(MWt)	2830	2914
Pressurizer pressure(psia)	2250	2250
Thermal design flow rare(gpm/loop)	92,600	92,600
RCS flow rate(lbm/s)	28,121	28,121
Core bypass rate(%)	6.8	6.8
Hot leg temperature(°F)	624.9	626.8
Cold leg temperature(°F)	556.4	556.6
Feedwater flow rate(lbm/s)	3559.3	3559.3
Feedwater temperature(°F)	440	440
Secondary side pressure(psia)	980	980
Steam flow rate(lbm/s)	1186.7	1221.3

Table 1 Initial Conditions Before and After Power Uprate

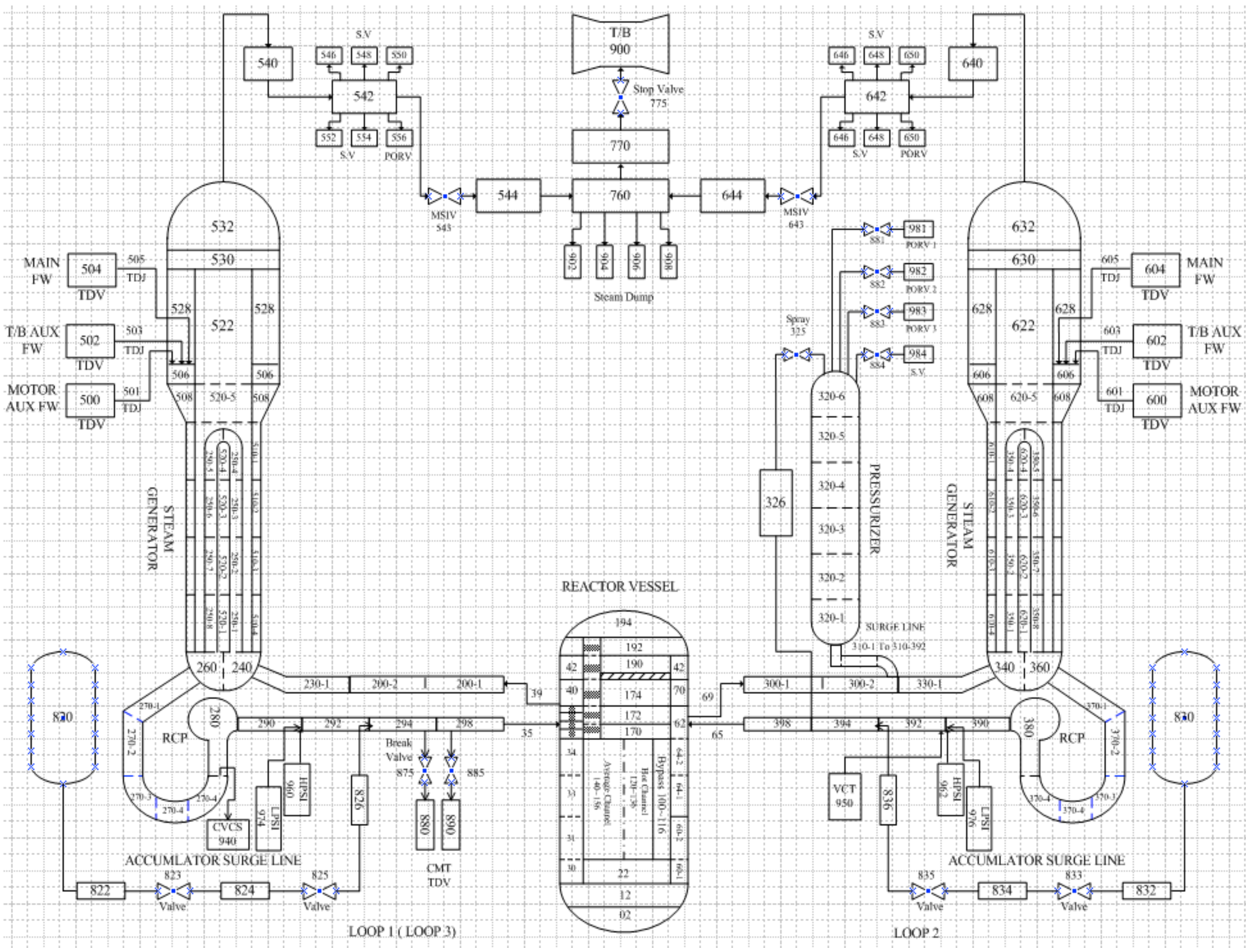


Figure 1 RELAP5-3D/K nodalization diagram of Maanshan NPP

2. Quantification of Safety Margin

To obtain the safety margin of a particular initiation event, following equation is used in SMAP to combine the PSA sequence frequencies and the safety margin of each individual sequence. In the equation, small break loss of coolant accident (SBLOCA) is used as the initiating event.

$$SM_{SBLOCA} = \sum_{i=1}^{n} \frac{freq_{Es\_i}}{freq_{SBLOCA}} SM_{Es\_i}$$

Where $freq_{SBLOCA}$ is initiating event frequency of small-break LOCA, $freq_{Es\_i}$ is frequency of accident scenario i, and SM_{Es_i} is the margin to the safety limit of accident scenario i as calculated in following equation. The second term in denominator represents the standard deviation for safety limit. The safety limit is a single value leads its standard deviation to be 0.

$$SM_{Es\_i} = \frac{\text{safety limit - best estimate value}}{\sqrt{\sigma^2_{best\ \text{estimate}} + 0^2}}$$

Conceptually, the different accident scenarios can be viewed as the source of aleatory uncertainty which reflects the performance of the system and its components. SM_{Es_i} is the epistemic uncertainty caused by the "imperfect knowledge" regarding values of input parameters of computational model and computational model itself. The ways to quantify these two uncertainties are described in the following paragraphs. In the present analysis, the safety margin of each individual accident scenarios is calculated using the 95th percentile with 95 % of confidence interval peak cladding temperature (PCT 95/95) during the accident.

2.1 Aleatory Uncertainty and PSA method

In the methodology of Probabilistic Safety Assessment of nuclear power plants, fault trees and event trees are adopted to identify the accident scenarios that could lead to core melt accident and to quantify their frequency. Figure 2 displays the event tree of small break loss of coolant accident of the plant analyzed. There are 12 headings in the event tree. The failure probability of each heading is marked on the figure. The end state of each branch is either OK or CD (core damage). For the accident scenarios that are classified as CD, the safety limit has already been exceeded. Therefore, the quantification of safety margin of these accident scenarios is not warranted. The safety margin quantification is required for the accident scenarios that are classified as OK (no core damage). As shown in Figure 2, these accident scenarios are S01, S02, S04, S07, S11 and S12.

In the present work, only one accident scenario is analyzed to demonstrate the uncertainty quantification methodology described in the follow sections. Accident scenario S01 is chosen for its high frequency and the transient response is relatively not complicated.

As shown in Figure 2, the frequency of S01 is 5.66E-003. The characteristics of the accident scenarios are loss of offsite power and reactor coolant pumps trip upon the initiation of the accident, all ECCS systems such as HPSI (high pressure safety injection), accumulators and LPSI (low pressure safety injection) are available.

2.2 Epistemic uncertainty and PIRT table

In the conventional uncertainty quantification of licensing calculations, best estimated thermal hydraulic code is used in the analysis. The code has to be validated against various separated effect and integral tests. The input values of parameters related to the modeling and plant operating conditions are tested for their sensitivity to the final results. The Phenomenon Identification and Ranking Table (PIRT) is built based on the results of sensitivity analysis. The probability density function (pdf) of parameters in PIRT is specified and the input values of these parameters are determined by Monte Carlo Sampling. The code calculations are repeated for a number of times with input vectors (combinations of input values of parameters in PIRT) from the sampling. The Best Estimate Plus Uncertainty (BEPU) methodology [1] adopts a slightly different approach in quantifying the uncertainty. In the approach, the Evaluation Model code is used in the calculations. The computational models in the code satisfy the requirements specified in 10 CFR 50 Appendix K of LOCA analysis. In BEPU approach, PIRT only contains parameters related to plant operating conditions. The input value of modeling parameters follows the requirements in Appendix K.

Statistically, there are two ways to determine PCT 95/95. In the parameter approach, the output is assumed to have certain distribution, e.g. normal distribution and the 95th upper bound value with 95% of confidence is:

$$X_{95/95} = \mu_{p,95\%} + 1.645\sigma_{p,95\%}$$

We can have sampling mean (μ_s) and sampling standard deviation (σ_s) by the results of sampling cases. Then use the following equation to get the population mean (μ_p) and population standard deviation (σ_p) for a upper bounding value of 95%.

$$\mu_p \le \left\lfloor \mu_s + t_\alpha (n-1) * \sigma_s / \sqrt{n} \right\rfloor$$

$$\sigma_p^2 \le \frac{\sigma_s^2 (n-1)}{\chi_{1-\alpha}^2 (n-1)}$$

Where t is student t distribution, χ^2 is chi-square, α equals 1 – confidence level, it will be 5% for 95% upper bounding and n is the sampling size.

In the non-parametric approach, the distribution of the output values is not a prior knowledge. For a specified confidence level (β), (γ)th percentile upper bounding value, when (N) number of calculations are made, the highest value of these results are the bounding values. The relation between sampling number (N), confidence (β) and PCT upper bounding value (γ) are:

$$\beta = 1-\gamma^N$$

For a 95% confidence and 95% upper bounding value, sampling number is 59. The highest value of 59 results is the upper 95th bounding value of 95% confidence level.

The Phenomenon Identification and Ranking Table (PIRT) used in the present analysis is shown in Table 2. The parameters listed in the table are considered to be the parameters that have the largest impact on the predicted peak cladding temperature during SBLOCA. The distribution of these parameters are obtained from Reference [2]. All of the PIRT parameters can be directly modified in the RELAP5-3D/K code, except initial fluid average temperature. Change the pressure boundary of the secondary side to modify it.

In the present analysis, two sets of calculations are made. The power in each set of calculations is 102% and 105% of rated power, respectively. The methodology of Reference [3] is adopted in the sampling the input values of the parameters in PIRT. The same 59 sample sets are used for both analyses.

In each calculation of RELA5-3D/K, the code is running with constant power up to 2000 seconds to set up the initial thermal hydraulic conditions of the calculations. After 2000 s calculation, the PIRT parameters are well modified. Replace the power calculation from a constant value to the point kinetic model of the code for another 500 seconds. Point kinetic model is used for neutron behavior calculation in SBLOCA transient. The actual SBLOCA transient is initiated at

2500 s and the calculation ends at 4000 s.

3. Results and Discussion

3.1 Transient Behavior of Reactor Coolant System in SBLOCA

In the SBLOCA analyzed, coolant rushes out from the system via the break of 3 in. diameter and reactor scram occurs almost immediately after as SBLOCA happens. At about 100 seconds into the transient, boiling starts and the void fraction in the tube side of SG increases. The heat transfer rate within the SGs. reduced. The pressure of the system is determined based on the energy balance between the decay heat generated, the energy loss from the break and heat removed from SG secondary side. The flow of high pressure injection system starts around 40 s. As RPV pressure drops below the set point the injection of accumulator starts. Break flow rate, ECCS injection, boiling in RPV and clearance of loop seal determine the water level in RPV. If the water level can be maintained above the top of active fuel during the transient, the temperature of cladding will not increase and PCT is lower than the temperature before the initiation of the accident. When the water drops below the top of active fuel, the core is uncovered and the cladding temperature rises above the temperature during normal operation. Based on the results of the present study, the transient behaviours of PCT during SBLOCA can be classified into three types. The timing of the clearance of loop seal plays an important role in the classifications of transient behaviours.

In the first type, the cladding temperature stays low in the transient due to core water level maintains above 60%. Figures 3 and 4 show the typical behaviours of PCT and normalized core water level of the first type. In the second type, the temperature of cladding reaches its peak value when the normalized water level is close to 50%. The value is lower than the initial cladding temperature. The behaviours of cladding temperature and normalized water level are displayed in Figures 5 and 6. In the third type, the transient cladding temperature is higher than the its initial temperature due to that the normalized core water level drops below 50%. The behaviours are shown in Figures 7 and 8.

The pressure difference between the inlet and outlet of steam generator U-tubes holds the liquid inside the U-tubes. The void fraction in the hot leg is higher than that in the cold leg. As the transient proceeds, the steam generated in the core pushes up the pressure in RPV higher. The pressure in hot leg increases and the pressure difference between inlet and outlet of SG U-tubes decreases to a point that cannot hold the liquid. Those liquid fall into the outlet plenum of S,G. and flow into reactor vessel via cold leg. The phenomenon is termed loop seal clearance [4]. Upon the clearance of loop seal, the water level in RPV recovers quickly and cladding is quenched. Loop seal clearance occurs both in the second and the third type. In the transient displayed in Figures 7 and 8, the clearance of loop seal occurs at 3580 seconds. Figure 9 shows the flow rate at outlet of U-tube of the same case.

Parameter	Distribution	Min	Max

Initial fluid average temperature, T_{avg} (°F)	Uniform	583.8	591.8
Pressurizer pressure, P_{RCS} (psia)	Uniform	2200.0	2300.0
Peak Heat Flux Hot Channel Factor, F_Q	Uniform (2.137±0.137) & Normal (σ = 2.6%)	2.000-4σ	2.274+4σ
Peak Hot Rod Enthalpy Rise Hot Channel Factor, $F_{\triangle H}$	Normal (Mean = 1.65, σ =1.9%)	Mean-4σ	Mean+4σ
Axial Power Distribution, P_{BOT}	Uniform	0.22	0.44
Axial Power Distribution, P_{MID}	Uniform	0.31	0.43
HPCI temperature (°F)	Uniform	49.0	120.0
Accumulator temperature (°F)	Uniform	100.0	150.0
Accumulator pressure (psia)	Uniform	632.0	680.0
Accumulator liquid volume (ft^3)	Uniform	985.0	1015.0

Table 2 PIRT for SBLOCA

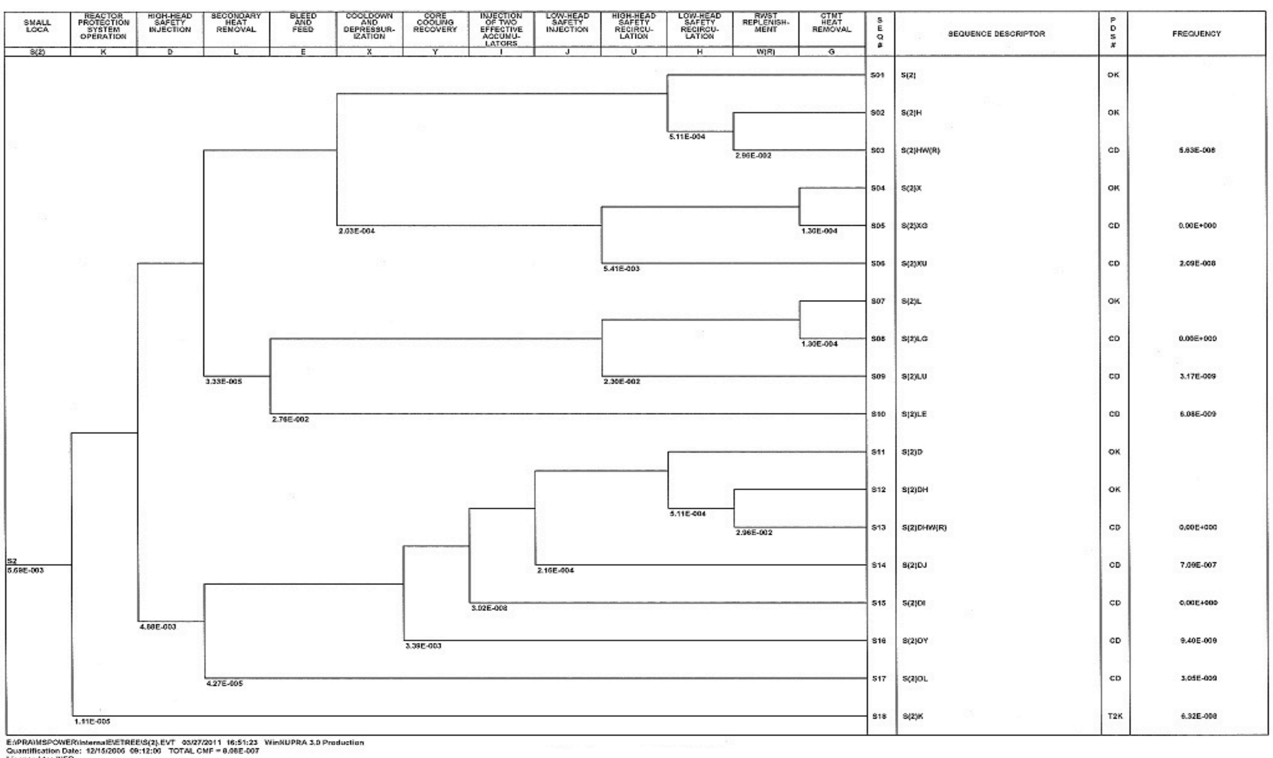


Figure 2 Event Tree for SBLOCA

3.2 PCT 95/95 and Distribution of Peak Cladding Temperature

The results of non-parametric approach show that the PCTs 95/95 values are 723.32°F and 714.55°F for 102% and 105% power, respectively. The highest PCT of each power level are from different sampling cases. For the 59 cases analysed for each power level, there are only two cases with predicted PCT higher than the initial cladding temperature, i.e. results of type 3. Based on these results, it can be concluded that, statistically, the safety margin for this particular accident scenario is not reduced due to power uprate from 102% to 105% power.

Figures 10 and 11 show the histogram for PCT of SBLOCA at 102% and 105% power, respectively. The sampling mean and standard deviation is 673.82°F and 8.396°F for 102% power, 674.31°F and 7.297°F for 105%. Calculations using Chi-square test have demonstrated that the distribution can be represented by normal distribution. Figure 12 displays the probability density function and the distribution of the output is assumed to be normal. For parametric approach: $X_{95/95} = \mu_{p,95\%} + 1.645 * \sigma_{p,95\%}$, the values are 684.11°F and 682.32°F, respectively for 102% and 105% power. As shown in Figure 12, the distribution becomes narrower after power uprate. The most probable PCTs before and after power rate are 675.89°F and 676.11°F, respectively. Table 3 summarized these results.

For this particular accident scenario, the non-parametric approach predicts higher PCTs than the parametric approach before and after power uprate. It implies that the non-parametric approach is more conservative than the parametric approach.

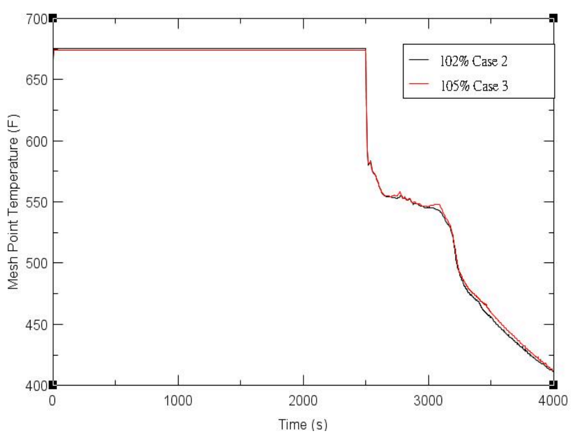


Figure 3 Type 1 Behaviors of Peak Cladding Temperature

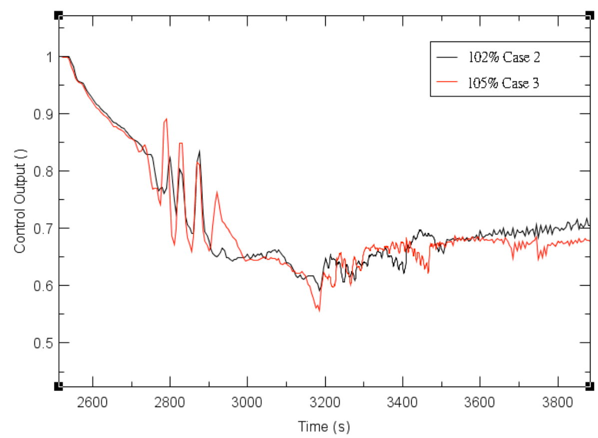


Figure 4 Type 1 Behaviors of Nominalized Core Water Level

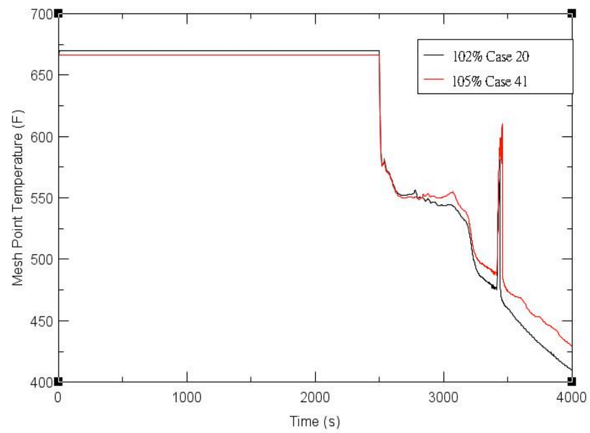


Figure 5 Type 2 Behaviors of Peak Cladding Temperature

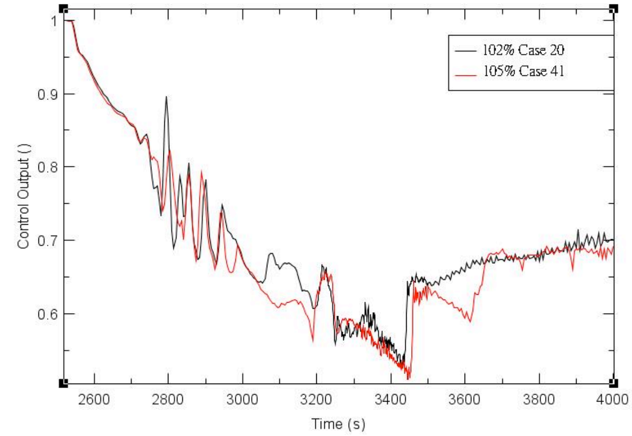


Figure 6 Type 2 Behaviors of Nominalized Core Water Level

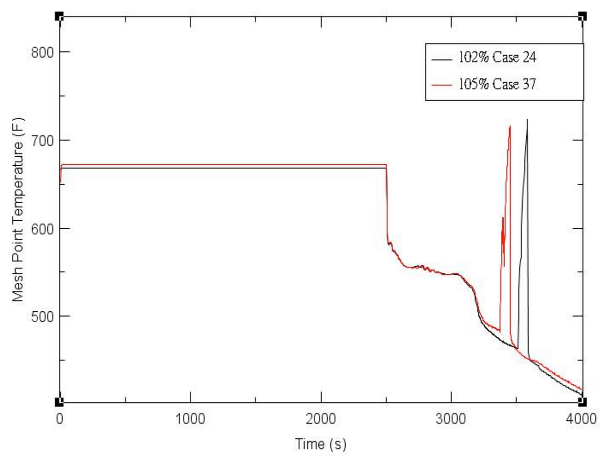


Figure 7 Type 3 Behaviors of Peak Cladding Temperature

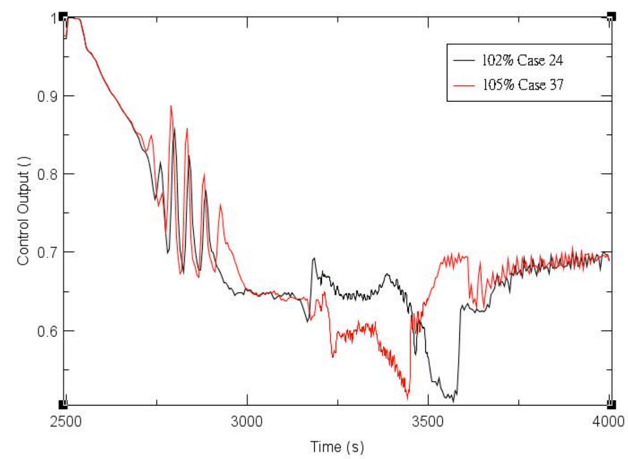


Figure 8 Type 3 Behaviors of Nominalized Core Water Level

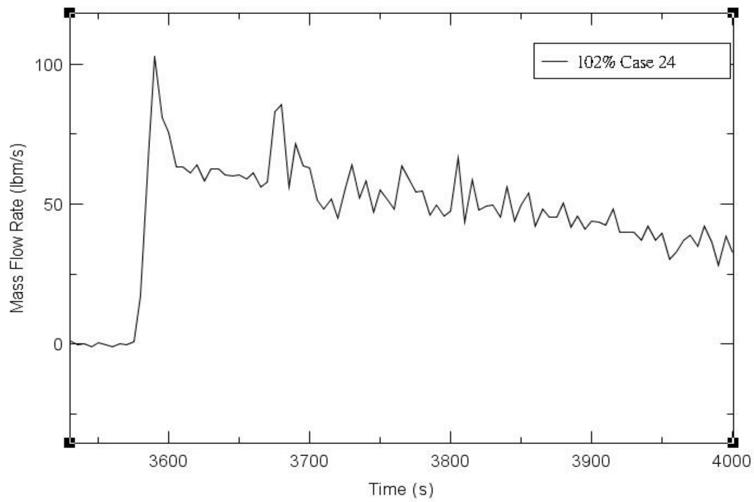
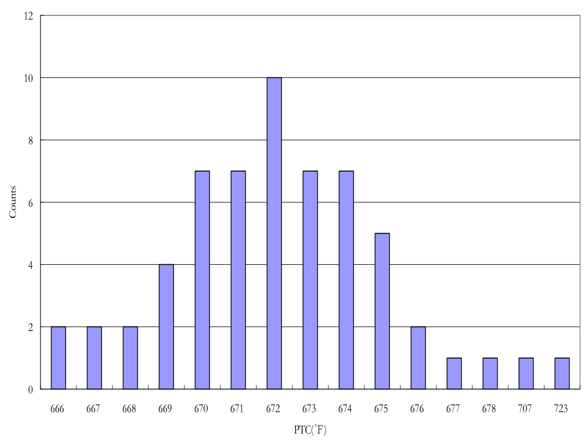


Figure 9 Flow rate at U-tube outlet



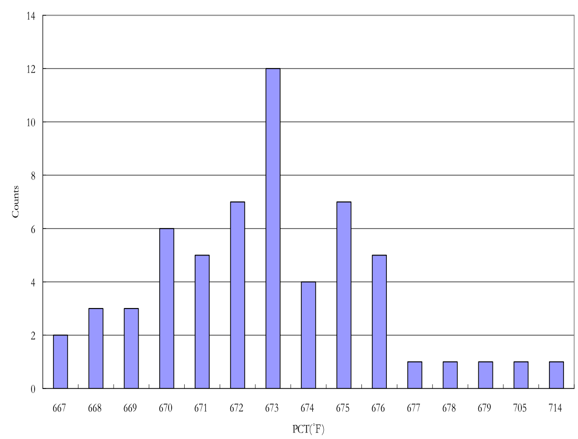


Figure 10 Histogram for PCT of SBLOCA at 102% power

Figure 11 Histogram for PCT of SBLOCA at 105% power

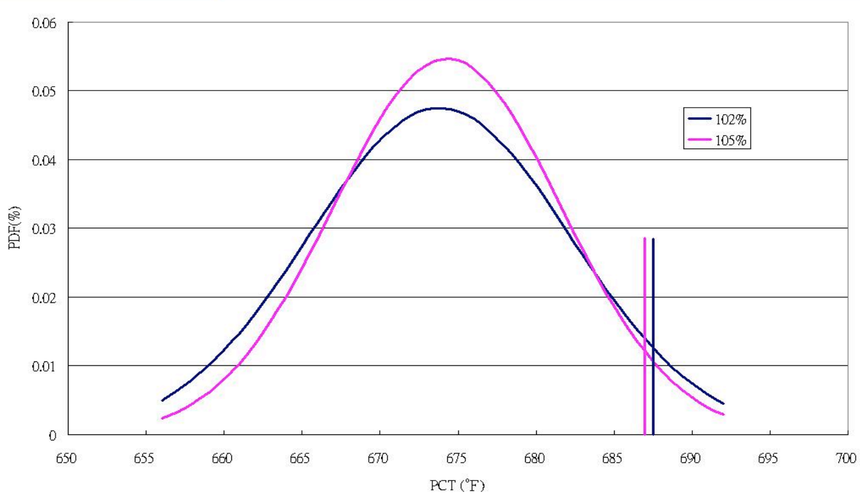


Figure 12 Probability Density Function of PCT and the 95% Upper Bounding Line

	102% MUR	105% SPU
Non-parametric	723.32°F	714.55°F
Parametric	684.11°F	682.32°F

Table 3 PCT in two methods

4. Conclusions and Future Work

In the present study, the SMAP methodology is adopted to quantify the reduction of safety margin of a 5% power uprate of Maanshan nuclear power plant of Taiwan Power Company. The sequence selected to demonstrate the methodology is small break cold leg loss of coolant accident. An evaluation model code, RELAP5-3D/K is used in the LOCA analysis. The plant specific PSA model is used to determine the accident scenario analysed. Uncertainty of the predicted PCT during the accident is quantified using the rank statistics. Phenomenon Identification and Ranking Table (PIRT) for parameters that are important for the small break LOCA analyses and their distribution are identified. Two sets of calculations are performed to determine PCT of the 95th percentile with 95% of confidence at 102% and 105% power cases. The results demonstrate that, statistically, the safety margin for the particular accident scenario analysed is not reduced due to power uprate from 102% to 105% power.

Based on the event tree of SBLOCA of the plant anaylzed, there are 6 accident scenarios need to analyse to quantify the reduction of safety margin due to power uprate. These calculations are the future work of the present study.

References

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Nomenclature

BEPU	Best Estimate Plus Uncertainty
CD	Core Damage
DOE	Department of Energy
HPSI	High Pressure Safety Injection
INL	Idaho National Lab
LPSI	Low Pressure Safety Injection
LWR	Light Water Reactor
MUR	Measurement Uncertainty Recapture
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
PCT 95/95	95th Percentile with 95 % of Confidence Interval Peak Cladding Temperature
PDF	Probability Density Function
PIRT	Phenomenon Identification and Ranking Table
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
SBLOCA	Small Break Loss Of Coolant Accident
SG	Steam Generators
SMAP	Safety Margin Action Plan
SPU	Stretch Power Uprate