# DETERMINATION OF REALISTIC MARGINS TO FUEL HEATUP FOR SLOW LOSS OF REGULATION ACCIDENTS IN A CANDU REACTOR

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## Abstract

The primary safety objectives related to Loss of Regulation (LOR) accidents are meeting acceptable dose rates and heat transport system overpressure protection limits. Within this context, the licensing requirements for operating CANDU reactors (e.g., R-8) require the prevention of fuel and pressure tube failures. To further simplify these acceptance criteria, the regulatory body in Canada has adopted regulatory guide G-144 which states that the first trip parameter should activate prior to dryout and the second trip parameter should limit post dryout temperatures to within 600 °C for less than 60s. For slow LOR events a probabilistic approach has been applied to demonstrate trip before dryout on both shutdown systems with high probability. However, CANDU reactor designs operate at much lower heat fluxes than Light Water Reactor (LWR) designs and hence the use of dryout (or in the LWR case, Departure from Nucleate Boiling) as an acceptance criteria is excessively conservative since the sheath and fuel temperature excursions in the post-dryout regime are benign. It has also been shown experimentally that operation in the post dryout regime for long durations is acceptable provided that the integrated effects of time and temperature are accounted for. This important design feature of a CANDU has thus far not been credited in LOR safety analysis. This paper presents the results of best estimate analysis of the margin to fuel failure during a slow Loss of Regulation accident accounting for the extremal nature of the accident analyses and the actual margins to fuel safety.

## 1. Introduction

In existing and new nuclear power plants a variety of special safety systems are employed which will trigger fast reactor shutdown in the event of an accident or undesirable plant condition. These special safety systems utilize multiple and redundant measurements of certain process and neutronic variables, known as trip parameters, which are continuously monitored against pre-determined limits. If a measured trip parameter deviates in an unsafe direction in excess of these pre-determined limits, known as trip setpoints, the special safety system will initiate a fast reactor shutdown. Nuclear Safety Analysis is performed to determine the plant response to hypothetical accident scenarios and to assess the effectiveness of the trip parameters and setpoints in achieving the safety goals (i.e., precluding fuel failures or minimizing public dose). Hence, Nuclear Safety Analysis is a critical component in the operation and regulatory licensing of nuclear power plants.

Changes in the regulatory framework for operating reactors are also driving changes in the methodology used to demonstrate plant safety [1]. Furthermore, Risk Informed Decision (RID) making practices and maintenance optimization [2] at each plant rely on accurate quantification of the impact of upgrades/refurbishment on safety margins. The Canadian Nuclear Safety

Commission (CNSC) and the USNRC have recognised that best estimate predictions of plant response, along with accurate assessments of uncertainties, is an acceptable alternative to more limiting and bounding analyses for demonstrating safety system response [3, 4]. The Canadian CANDU industry is currently pursuing the use of Best Estimate and Uncertainty (BEAU) methodologies to resolve various issues related to Loss of Power Regulation, Loss of Coolant and Loss of Station Power Accidents [5]. Extensions of best estimate methodologies for Neutron Overpower Protection and maximum channel power compliance have been studied by Sermer et al. [6, 7].

The treatment of uncertainties which arise from imperfect knowledge of the phenomena related to the accidents results in epistemic uncertainties in the trip setpoint predictions. In the past, deterministic analysis methodologies involving bounding assumptions were used to ensure adequate plant response for postulated accident scenarios. As a result of these simplifications it is impossible to determine the exact margins to safety limits. An analysis of the effects of these epistemic uncertainties using Extreme Value Statistics is available in Reference 8. Furthermore, due to scientific discovery issues combined with plant safety margin deterioration due to component aging these traditional methodologies predict consequences which may challenge full power operation. Finally, operational changes that occur periodically (e.g., online fuelling, changes in inlet temperature with time due to aging) further complicate the methodologies since the required trip setpoint may vary as a function of time. These changes lead to aleatory uncertainties in final trip setpoint analyses and must be factored into the setpoint methodology. Application of a two-tiered approach for epistemic and aleatory uncertainties is provided in Reference 9 for accidents involving loss of feedwater and stressed the importance of detailed uncertainty propagation methodologies.

This paper expands on the updated methodologies discussed in the References and examines the total margins to safety for a slow Loss of Regulation (LOR) accident. In particular, this paper examines the impact of the probability and confidence limits on the trip setpoint and their associated impact on significant fuel overheating and potential fuel damage for a slow LOR event.

## 2. Methodology

For a typical CANDU reactor there are approximately 480 fuel channel assemblies in the reactor core which are fed by two separate figure-of-eight heat transport system loops. Each figure-of-eight loop has 2 heat transport system pumps and 2 steam generators for heat removal and provides coolant flow to half of the fuel channels. The 480 fuel channels contain 12 to 13 natural Uranium fuel bundles at power levels up to approximately 6 MW per channel. Each fuel channel undergoes on-line fuelling at different points in time in an operating cycle and hence the actual fuel channel powers are a function of time. Furthermore, as a result of fuelling or other day-to-day operating variances, the heat transport system conditions (flows, temperatures and pressures) also vary to some extent. Since there are variations in the reactor physics parameters (e.g., fuel channel power due to fuelling, in-core detector drift, and total reactor power) and thermalhydraulic conditions, determination of accident behavior is more complicated than if the reactor conditions were completely fixed in time. These changes lead to aleatory uncertainties in final trip setpoint analyses and must be factored into the setpoint methodology.

CANDU reactors are equipped with two independent Shutdown Systems, each with the capability of rendering the core subcritical and each with its own unique set of instrumentation. The instrumentation systems within each Shutdown System are divided into three logic channels and within each logic channel there are several redundant instruments measuring plant variables. The shutoff mechanism relays are actuated when trip signals from two-out-of-three exceed their trip setpoint. In the event of an accident at a CANDU station, the transients may be terminated by the RRS monitoring systems or either of the special safety shutdown systems. For this work, the Neutron Overpower Protection (NOP) system is examined with respect to its capability to mitigate a Loss of Power Regulation event. The typical CANDU NOP system consists of 40 to 50 self-power in-core flux detectors arranged into three separate logic channels. In the event of an abnormal power transient, a rector trip is initiated when 2-out-of-3 trip logic vote, with a given logic system voting when a detector signal exceeds some pre-determined installed setpoint.

The methodology outlined below is based on that documented in References 7 and 9 which is often referred to as Extreme Value Statistics (EVS). The EVS approach contains the following unique characteristics as compared to other Best Estimate and Uncertainty methodologies:

1. It propagates aleatory and epistemic uncertainties separately which allows for a two-tiered approach for uncertainty analysis.
2. It propagates uncertainties through all instrumentation and fuel channels simultaneously, and hence more accurately determines the probability and confidence levels as compared to other approaches.

As noted in the references, these features of the EVS methodology come at a significantly higher computational burden; however, with the application and utilization of massively distributed computing techniques at McMaster University fully two-tiered and multiple pathway uncertainty propagation methods are possible. The methodology outlined below utilizes these features, while the remaining paper examines the inherent conservatisms within the derived acceptance criteria utilized in the analysis and the resultant impact of the 95-95 acceptance criterion adopted in the methodology. The following subsections are intended to provide a summary of the methodologies discussed in References 7 and 9.

### 2.1 Epistemic uncertainty analysis methodology

Reference 9 provided the methodology used to determine the safety system trip setpoint epistemic uncertainty based:

- initial operating conditions uncertainty,
- simulation uncertainty, and
- safety system instrument uncertainties.

The methodology was used to define an instantaneous Required Trip Setpoint (*RTSP*) which can be computed probabilistically. *RTSP* was defined as the setpoint required to initiate a reactor shutdown during a given postulated accident such that there is high probability that the acceptance criteria will be met at an instantaneous reactor configuration, *m*. The *RTSP* for state *m* must account for: 1. the epistemic uncertainty in instantaneous plant boundary conditions, 2. the uncertainty in simulation models and computer codes used to predict the plant response, 3.

the measurement uncertainties related to shutdown system instrumentation, and 4. the instrument time delays and uncertainties in time delay if necessary. Finally the aleatory uncertainties cause by transitions from state-to-state must be included.

The true trip setpoint for an instantaneous reactor state, $tsp_m$, is defined as the setpoint required to meet the acceptance criterion given complete knowledge of the initial plant conditions at that instant, perfect computational models for that accident sequence and perfect measurements. Essentially the trip setpoint can be defined as:

$$tsp_m = \min_{1...a}\left[mtd_i\right] \times \min_{d,e,f}\left(\max_{1...b}\left[dr_f\right]\right) \qquad (1)$$

where $mtd_i$ is the margin to dryout for each fuel channel i, $dr_j$ is the detector reading maximized over all detectors in a logic channel, and *d, e, f* indicate each of the logic channels in a three-out-of-there logic channel system. Note that the margin to dryout for a channel is its ratio of the critical channel power to the channel power. Since the conditions, models and measurements are not perfectly known, only an estimate of the setpoint, $TSP_m$ is available and is computed from the estimated margin to dryout and detector readings as follows:

$$tsp_m = \min_{1...a}\left[mtd_i\right] \times \min_{d,e,f}\left(\max_{1...b}\left[dr_f\right]\right) \qquad (2)$$

Note the nomenclature is such that lower case variables indicated an exact quantity while upper case variables indicate an estimate. The relationship between this estimate and true value is given as:

$$TSP_m = tsp_m\left(1 + \varepsilon_m^{tsp}\right) \qquad (3)$$

where $\varepsilon_m$ is the error in the estimated setpoint at that instant in time and is a random variable which considers epistemic uncertainties in the initial conditions, plant response models and instrumentation. Consequently $TSP_m$ is a random variable. In order to obtain the required/installed setpoint, *RTSP*, we must ensure that the minimum margin to dryout remain above 1.0 during all times considered in the reactor transient, i.e,: $mmtd_m > 1.0$ as computed from:

$$mmtd_m = \min_{1...a}\left[\frac{ccp_{i,m}}{cp_{i,m}}\right] \qquad (4)$$

for an initial reactor state *m* and where $cp_i$ is the instantaneous channel power and $ccp_i$ is defined as the Critical Channel Power in channel *i*. The Critical Channel Power (*CCP*) corresponds to the channel power that would be required to initiate dryout for the same thermalhydraulic inlet boundary conditions. However, the quantities above are subjected to epistemic uncertainties resulting from random prediction uncertainties as well as aleatory uncertainties resulting from plant condition variability. The estimated minimum margin to dryout in a given fuel channel *i*, $MTD_i$, is predicted by the computer codes and is related to the true margin to dryout by the following:

$$MTD_{i,m} = mtd_{i,m}\left(1+\varepsilon_{i,m}^{mtd}\right) \tag{5}$$

where $mtd_i$ is the true margin and $\varepsilon^{mtd}$ denotes the error in determination of the fuel channel *i* margin to dryout.

Based on the simulated minimum margin to dryout, *MMTD*, a simulated reactor trip setpoint can be determined (*TSP)*. For CANDU reactors the epistemic uncertainties in margin to dryout are determined by comparing models to full-scale experiments over the range of possible accident conditions. In a similar fashion, the detector response within each safety system logic channel can also be considered. Typical values of the epistemic uncertainty in MMTD and DR for a slow Loss of Regulation event are approximately 5% and 1.5%, respectively. Since this trip setpoint is subjected to the epistemic errors in margin to dryout and detector response uncertainty, this *TSP* will be in error with respect to the true trip setpoint, tsp:

$$\varepsilon_m^{tsp} = \frac{TSP_m - tsp_m}{tsp_m} \tag{6}$$

where $\varepsilon^{tsp}$ is the error in the estimated trip setpoint at reactor state *m*. A Monte-Carlo method is then used to assess this error distribution based on parental epistemic uncertainties in the setpoint calculation. The epistemic parental uncertainties considered in this work are those resulting from critical heat flux (CHF), single and two phase flow resistances, and simulation uncertainties related to the channel power (CP) and detector reading (DR) responses. For simplicity the parental uncertainties were assumed independent and normal, although any distribution and/or covariant effects could be included.

### 2.2 Aleatory uncertainty methodology

Considering Equation (1), the aleatory uncertainties related to both the minimum margin to dryout and the flux detector readings must be considered. With respect to the margin to dryout the relevant aleatory components are:

- the power in each individual fuel channel at the time of the event which varies based on physical online fuelling during operation and fuel burn-up. The dryout power which depends on the reactor temperatures, pressures, flow losses which are may be a function of time due to aging, plant changes (e.g., feedpump configuration and/or axial power variation due to burnup) and fuelling.

With respect to the detector responses during an event, the value of the response will be a function of the initial detector readings at the time of the accident which in turn result from:

- fuelling operations which can affect nearby detector readings as well as cause changes in reactivity device positions (i.e., liquid zone controller levels in the core).

- instrument and transmitter detector fluctuations and drift as well as calibration which is periodically performed during operation[1].

In order to assess the aleatory uncertainties physical plant data was collected from several operating CANDU units for each of the aleatory sources over a 3 year operating period. Data collection was synchronized such that for each reactor state, *m*, a complete set of data is obtained for MMTD-FUEL, MMTD-OPS, DR-FUEL and DR-OPS. Details on the treatment of aleatory uncertainties based on station operations is provided in Reference 9.

For each of these defined reactor states, the epistemic uncertainty analysis described above is repeated and hence the methodology can be considered a two-tiered approach where the inner loop examines the epistemic uncertainties for a given state, and the outer-loop assess the aleatory contributions. Once the entire process is completed for all core states an installed setpoint *itsp* can be determined based on a suitable lower bound which satisfies the 95%-probability and 95%-confidence approach defined in ISA 67.04 as shown in References 8.

## 3. Acceptance Criteria Assessment

The derived acceptance criteria of trip before dryout is a sufficient, but not necessary, criterion for ensuring fuel and fuel channel integrity, and hence ensures dose limits are met. For slow LOR events, the methodology above may be used to determine the setpoint required to ensure the prevention of dryout with 95%-probability and with 95%-confidence over time. It should be noted that the above methodology can be employed for any combination of probability and confidence interval specifications (e.g. 95/95, 98/95, 98/90). But the question than arises as to the acceptability of a 95/95 criterion and the subsequent accident consequences should these limits be exceeded during an actual LOR event. Within the context of accident analysis, it must be noted that the consequence of having a setpoint which is slightly higher than that needed to prevent dryout for a given core configuration does not have significant impact on core or fuel damage risk due to:

- The numerous STEPBACK parameters are available for such events. The STEPBACK may either cause power reductions early in the event and hence precede SDS action, or they may occur after SDS action due to the higher number of detectors (and hence higher probability of trip) for the NOP/ROP system.
- The probability of complete NOP SDS system failure is very low due to the large number of detectors and detector redundancy in the core. Furthermore the redundant coverage from the completely separate SDS systems ensures eventual reactor shutdown.
- The analysis methodology ignores the lag between neutronic power and power to coolant.
- In the event where an NOP trip is late (post dryout) for an LOR event, sufficient fuel cooling is still available from the HTS system in the interim until the power increase is such that the system does activate. While there is a small but finite probability that a trip may occur after dryout, the consequences in terms of fuel heat up are very benign (i.e., there are no cliff edge effects in terms of fuel failure mechanisms). The actual risk of fuel failure by a “late” NOP trip for a slow LOR event is very small. The objective of this work is to quantify the probability of significant fuel heat-up for limiting core states.

[1] Note that depending on detector location in the core each detector will have different variability due to its physical location (e.g., proximity to reactivity devices, frequency of required fuelling in that region, etc..). The detector specific variability measured was retained and applied in the analysis.

To satisfy the questions related to the consequences of falling outside of the statistical coverage determined this paper specifically:

1. determines a trip setpoint required to prevent dryout using the 95/95 criteria specified in literature for a generic CANDU plant for one of the two separate and diverse trip setpoints and for a given set of parental uncertainties
2. determines the probability of trip before 600 °C for the same setpoints and parental distributions as defined above.
3. determines the required trip setpoint needed to prevent 600 °C, and hence the ***overpower margin*** between the dryout prevention setpoint and the 600 °C setpoint at a consistent probability and confidence level.

The above methodology uses the alternate derived acceptance criteria of 600 C, and hence maintains a large margin to potential fuel failures. The Critical Channel Powers (CCPs) used in this work were determined using detailed fuel channel and feeder geometries for each fuel channel in the core with the NUCIRC computer code. For determining the overpower required to obtain 600C the same assumptions were employed as for the CCP calculations, but the post dryout sheath temperatures were determined using the Groeneveld-Delorme post dryout heat transfer correlation [10] with the following conservative (bounding) overpower bundle factor:

$$\left(\frac{\mathrm{h}}{\mathrm{h}_{\mathrm{GD}}}\right) = 1 + C_1 \exp(-C_2\ (\mathrm{OHFR} - 1)^{C_3}) \tag{7}$$

where $h_{GD}$ is the predicted heat transfer coefficient by Groeneveld-Delorme, OHFR is the Over Heat Flux Ratio which is the ratio of the local heat flux to the CHF, and $C_1$, $C_2$ and $C_3$ are empirical constants derived from the full scale 37-element Stern Laboratories PDO database. Since Equation (7) represents a bounding PDO heat transfer in 37-element bundles it should be noted that there are additional degrees of conservatism within this work, in addition to those discussed previously.

## 4. Results

The methodology described above was coded into MATLAB and the associated trip setpoints were established. For a randomly selected reactor core state, m, the estimated epistemic uncertainty distribution is shown in Figure 1. The overall characteristic and shape of the resultant uncertainty resembles a Weibull distribution as expected due to the extreme value statistics (EVS) nature of the trip setpoint calculation. For this particular reactor state the 95-percentile lower bound probability is selected such that there is a 95% probability of trip before dryout for this case. As mentioned in the preceding sections, this Monte-Carlo procedure was repeated for a set of 1000 randomly selected initial core configurations and the setpoint required for each state is shown in Figure 2. Finally, the overall 95-percentile lower bound over all reactor states is then selected in order to provide a 95% probability of trip before dryout over 95% of the available operating states. The trip setpoint in this case is 120.2%FP adjusted for calibration factors.

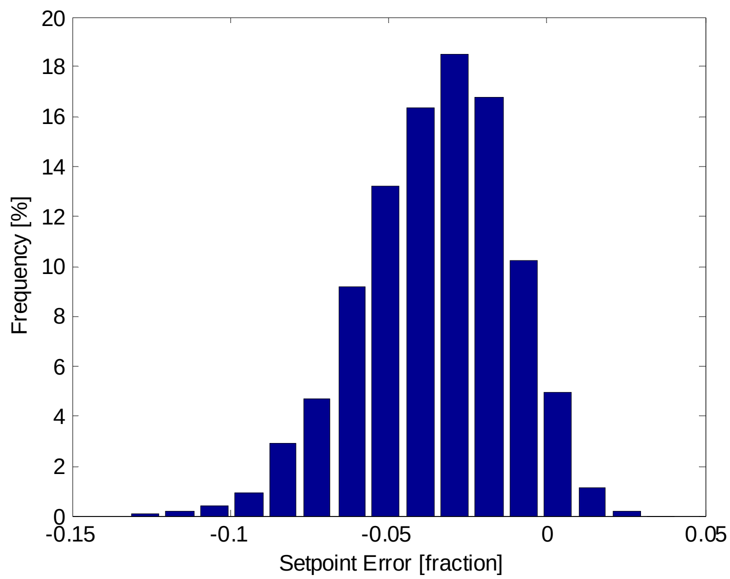


**Figure 1: Epistemic Uncertainty Distribution in Trip Setpoint for a Randomly Selected Initial Core State**

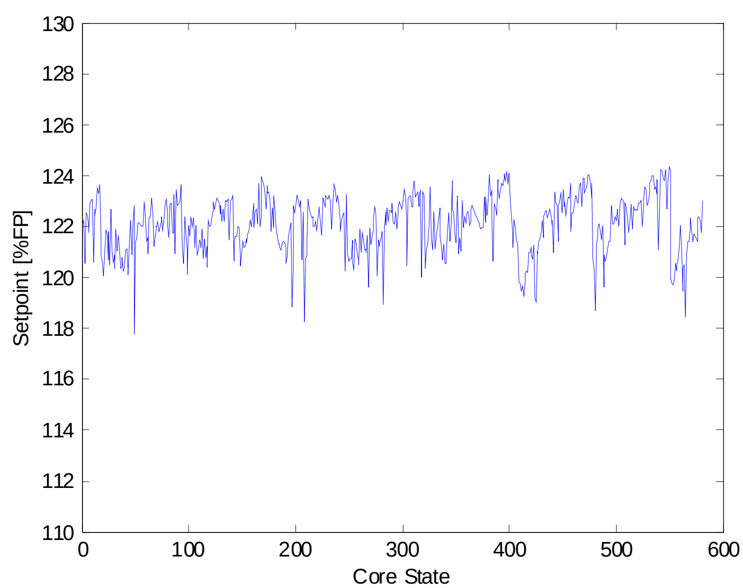


**Figure 2: Trip Setpoint Required to Prevent Dryout as a Function of Sampled Reactor State (effect of aleatory uncertainties)**

In order to determine the consequences of a delayed NOP trip and the small probability of dryout the uncertainty analysis is repeated using the PDO heat transfer correlations discussed above and an alternate derived acceptance criteria of 600 °C. NUCIRC was used to predict the Critical Channel Power based on Temperature (CCPT) using the same inlet flow conditions as for the CCP predictions. The average increase in the CCPT compared to the CCP was approximately 21 to 24% for the conditions analyzed and was correlated to initial channel flow. Using the CCPT instead of the CCP, the trip setpoint required to ensure prevention of 600 °C was estimated for the same initial conditions and fuel channel ripple set as those shown in Figures 1 and 2 and the results are shown in Figures 3 and 4. Figure 3 shows similar Weibull characteristic as Figure 1 while Figure 4 shows a substantive improvement in the setpoint as expected. The setpoint required to prevent 600 °C with 95% probability and 95% confidence was estimated from these results to be 144.3%FP.

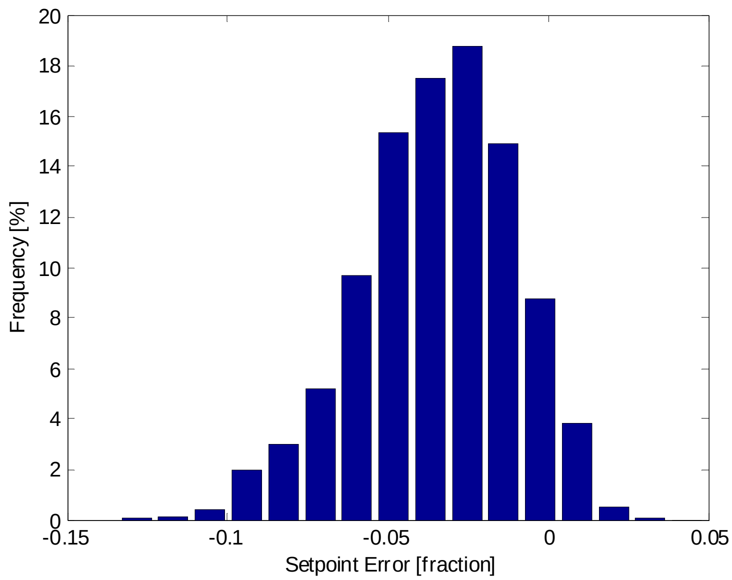


**Figure 3: Epistemic Uncertainty Distribution in Trip Setpoint for a Randomly Selected Initial Core State for a CCPT Calculation**

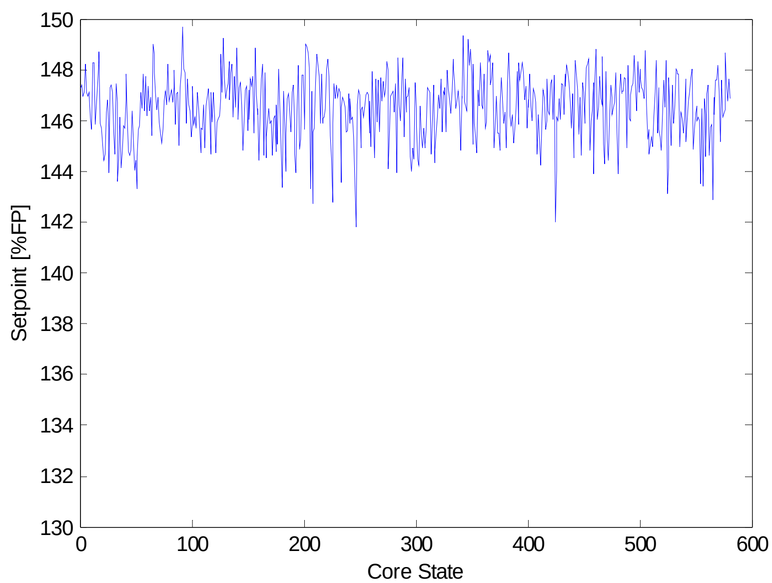


**Figure 4: Trip Setpoint Required to Prevent 600 °C as a Function of Sampled Reactor State**

The results can further be interpreted by examining the core states where a required trip setpoint of less than 120.2 %FP is required. For example, the error probability distribution for the CCPT of core state number 50 was examined as shown in Figure 5 which corresponds to the absolute lowest setpoint observed in Figure 2. Specifically, Figure 5 shows the distribution of reactor core powers[2] that may cause a channel to exceed 600 °C along with an indication of the installed trip setpoint based on the dryout criteria. Since the reactor will trip at an installed setpoint of 120.2 %FP, it can be seen from this figure that the probability of reaching a 600 °C for this tripping power is very small, even for the worst core state analyzed in this study. In other words, although core state 50 requires a trip setpoint of less than 118 %FP to prevent dryout, Figure 5 shows that the overall setpoint of 120.2 %FP derived in the previous section provides a very high assurance that significant fuel heat up will not occur, even for this worst case core state.

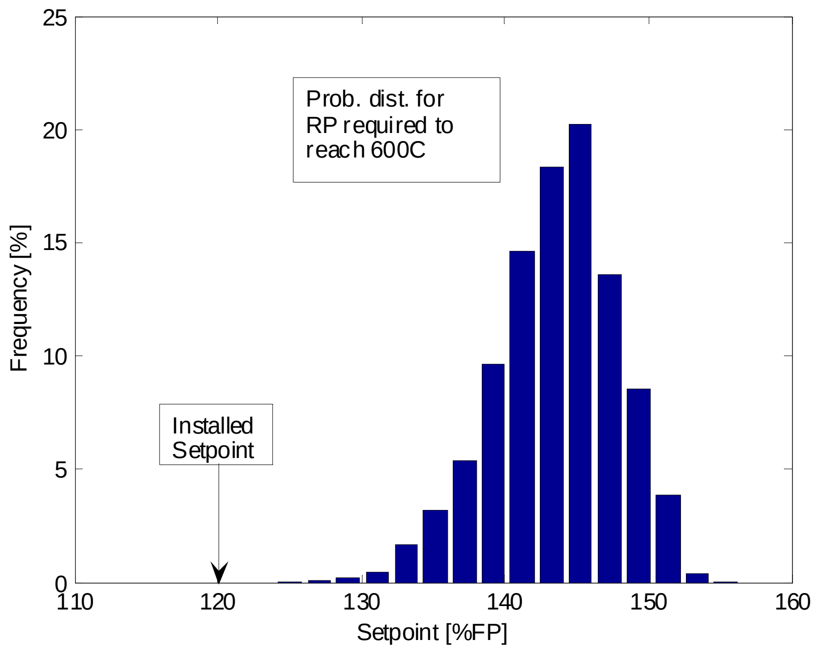


**Figure 5: Distribution of Estimated Setpoint Required to Prevent 600C in a Channel for a Limiting Core State (core state #50 having the absolute lowest required trip setpoint).**

To quantify the probability of exceeding 600 °C for this low probability core state further statistical analysis was performed on the data used to create the distribution shown in Figure 5. For this representative core state (core state #50 within the data presented previously) the estimated setpoints was fit to the following probability function within MATLAB:

[2] It should also be noted that the Reactor Powers have been reduced by a factor of 1.18 which was the calibration factor in effect over all the core states selected. Hence the Reactor Power shown is the real reactor power, not the power indicated by the detectors after calibration.

$$p = \left(\frac{1}{\sigma}\right)\left(1+\kappa\frac{(x-\mu)}{\sigma}\right)^{-1-\frac{1}{\kappa}} e^{-\left(1+\kappa\frac{(x-\mu)}{\sigma}\right)^{-\frac{1}{\kappa}}} \quad (8)$$

where μ, κ and σ are the location, shape and scale parameters for the general Weibull distribution. Using the fitted distribution we can further assess the consequences of an event that evolves from this low probability core state using methodologies similar to the extremal forecasting techniques used in the insurance industry (see for example Reference 11). In particular, for the fitted 3-parameter Weibull distribution it can be shown that this 120.2%FP setpoint, while it may not prevent dryout with the same probability as other core states, it does provide greater than 99.99% probability of preventing 600 °C fuel temperature. This can be interpreted as the probability of fuel sheath temperatures exceeding 600 °C for an installed trip setpoint of 120.2 %FP is less than 0.01 % for this state. Furthermore, the reactor power would need to exceed the installed trip setpoint of 120.2 by over 14%FP for there to be an appreciable probability of 600 C fuel temperatures (i.e., 14% FP beyond the trip setpoint is required to generate a 5% probability of a 600 C sheath temperature or above). All available core configurations with a required setpoint below the installed limit of 120.2%FP were similarly assessed and for every case there was a greater than 99.99% probability of a trip before 600 °C. This demonstrates that even for the very low probability core states where dryout cannot be precluded with a setpoint of 120.2%FP, there is an extremely low probability of fuel sheath temperature excursions above 600 °C.

## 5. Conclusions

The objective of this study was to realistically quantify the margins associated with the NOP trip using a robust statistical methodology and examining the post dryout heat up of the fuel. Within this context, it should first be noted that the following conservatisms were retained in this work:

- No attempt was made to use a best estimate PDO heat transfer correlation. Rather the bounding overpower bundle correlation methodology was used. Use of a best estimate PDO correlation would have shown much larger margins to fuel heat-up than shown in this work.
- STEPBACK initiation was conservatively not credited in the analysis. Since a STEPBACK is expected either shortly before, or soon after the SDS setpoint, the probability of power increases to the point of 600 °C would be further reduced.
- No credit is taken for the margin between the 600 °C criterion adopted, and the power required to initiate fuel failures.
- No credit is taken for the lag between neutronic power and power to coolant.

From the study documented in this paper it has been shown the EVS methodology, based on a two-tiered uncertainty propagation methodology, provides setpoints which meet a prevention of dryout acceptance criterion of 95%-probability with 95% confidence over the operating duration analyzed. Furthermore, by examining the worst sampled operating states it can be shown that the probability of reaching 600 °C with a setpoint based on this methodology is significantly less than 0.01% and hence there are very large margins to fuel failure for the data analyzed. Finally it should be noted that the above analysis was completed for the time averaged reactor core configuration and additional work is ongoing for off-normal flux shapes (e.g. adjuster usage).

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