Uncertainty Analysis of Suppression Pool Heating During an ATWS in a BWR-5 Plant

An Application of the CSAU Methodology Using the BNL Engineering Plant Analyzer

Manuscript Completed: February 1994
Date Published: March 1994

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**Prepared for
Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC FIN A3983**

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Abstract

The uncertainty has been estimated of predicting the peak temperature in the suppression pool of a BWR power plant, which undergoes an NRC-postulated Anticipated Transient Without Scram (ATWS).

The ATWS is initiated by recirculation-pump trips, and then leads to power and flow oscillations as they had occurred at the LaSalle-2 Power Station in March of 1988. After limit-cycle oscillations have been established, the turbines are tripped, but without MSIV closure, allowing steam discharge through the turbine bypass into the condenser. Postulated operator actions, namely to lower the reactor vessel pressure and the level elevation in the downcomer, are simulated by a robot model, which accounts for operator uncertainty. All balance of plant and control systems modeling uncertainties were part of the statistical uncertainty analysis that was patterned after the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology.

The analysis showed that the predicted suppression-pool peak temperature of 329.3 K (133°F) has a 95-percentile uncertainty of 14.4 K (26°F), and that the size of this uncertainty bracket is dominated by the experimental uncertainty of measuring Safety and Relief Valve mass flow rates under critical-flow conditions.

The analysis showed also that the probability of exceeding the suppression-pool temperature limit of 352.6 K (175°F) is most likely zero (it is estimated as $< 5 \cdot 10^{-4}$). The square root of the sum of the squares of all the computed peak pool temperatures is 350.7 K (171.6°F).

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Executive Summary

E.1 Objective

The statistical uncertainty analysis presented in this report is a follow-on to the BWR stability analysis carried out earlier [Wulff et al. 1991], subsequent to the power and flow oscillation event in the LaSalle-2 Boiling Water Reactor (BWR) on March 9, 1988.

One of fourteen NRC-stipulated questions raised the issue of whether or not "available computer codes are reliable for predicting BWR instability". The purpose of the work presented here is to resolve this issue for the BNL Engineering Plant Analyzer (EPA), simulating a specific, NRC-postulated Anticipated Transient Without Scram (ATWS).

In particular, the objective of the work is to quantify for the LaSalle BWR power plant, systematically and traceably, the uncertainty in simulating the rise of suppression-pool temperature during an ATWS with power and flow oscillations, and to estimate the probability that the pool temperature could exceed the specified 175°F temperature limit.

The work presented here supplements the previously performed EPA assessment (see [Wulff et al. 1991], Chapters 6, 7, and 9).

E.2 The BWR Scenario

The ATWS is initiated by recirculation-pump trips and leads to power and flow oscillations as they had occurred at the LaSalle-2 Power Station in March of 1988. After limit-cycle oscillations have been established, the turbines are tripped, but without MSIV closure, allowing steam discharge through the turbine bypass into the condenser. Postulated operator actions, namely actions to lower the reactor vessel pressure and the level elevation in the downcomer, are simulated by a robot model, which accounts for operator uncertainty. All balance of plant and control systems modeling uncertainties were part of the statistical uncertainty analysis.

E.3 Approach

The systematic and auditable Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology [Boyack et al. 1990] was used for guidance to quantify the simulation uncertainty resulting from the uncertainties introduced by modeling uncertainties and the specification of initial plant and boundary conditions.

Uncertainties arising from operator interventions, as postulated by the NRC, were introduced by varying key parameters of an operator-simulating robot model.

The Phenomenon Identification and Ranking process (part of CSAU), and the documentation of the HIPA (for High-Speed Interactive Plant Analyzer) code of the EPA lead to the identification of 19 key modeling parameters.

These 19 parameters were varied systematically in the range of their uncertainties, and the variations were used as input data for 63 full-plant simulations. The simulation results, i.e. the resulting suppression-pool peak temperature, served to subtend, through least-square fitting, a response surface, which approximates the EPA response to variation of the key modeling parameters.

Acknowledgements

The work documented in this report was performed under the auspices of the United States Nuclear Regulatory Commission (USNRC). It was funded by the Reactor and Plant Systems Branch (RPSB), of the Division of Systems Research (DSR), in the Office of Nuclear Regulatory Research (RES), under FIN Number A-3983. The program was monitored at the USNRC by Harold Scott, whose support of this work is greatly appreciated.

The work was carried out in cooperation with the Technical Review Group (TRG), whose TRG members were D. E. Bessette (USNRC-RES), S. Fabic (USNRC-AEOD), G. Johnsen (INEL), G. S. Lellouche (SLI), J. March-Leuba (ORNL), L. E. Phillips (USNRC-NRR), H. J. Richings (USNRC-NRR), U. S. Rohatgi (BNL), Z. Rouhani (INEL), W. Wulff (BNL) and N. Zuber (USNRC-RES). The guidance and support of the TRG members are also greatly appreciated.

The ranking of relevant phenomena was carried out by TRG in January of 1991, the Phenomena Identification and Ranking Table (PIRT) was developed in March 1991 by G. Johnsen of the Idaho National Laboratory (INEL), and the formal PIRT documentation (Section 2.4) was transmitted on diskette to Brookhaven National Laboratory (BNL) in September 1993.

The reference transient and forty-five variations for fitting the response surface were simulated by H. S. Cheng and A. N. Mallen on the BNL Engineering Plant Analyzer in May of 1991. Eighteen additional simulations were performed by W. Wulff at BNL in August of 1991 to resolve observed, but unexpected, anomalies in the previously completed simulations (Section 2.7.4). A first draft of this report, documenting the BNL work was transmitted to the NRC in June of 1992. W. Wulff prepared the final draft of this report. Mrs. N. Siemon accomplished the difficult task of editing and bringing this final report in compliance with NRC documentation requirements.

The statistical analysis presented in Section 2.8 and in Appendix A was carried out by G. S. Lellouche of Technical Data Services, Inc. (TDS). The final documentation of the statistical analysis was received from INEL by BNL in September 1993.

1 Introduction

1.1 ATWS Stability Issue for BWRs

Following the unexpected, but safely terminated, power and flow oscillations in the LaSalle-2 Boiling Water Reactor (BWR) on March 9, 1988, the Nuclear Regulatory Commission (NRC) Offices of Nuclear Reactor Regulation (NRR) and of Analysis and Evaluation of Operational Data (AEOD) concluded that BWR Stability research was needed to determine if unexpected conditions had occurred at the plant, if previously performed stability analyses were inadequate, or if power and flow oscillations could give rise to new safety issues.

NRR and AEOD requested that the Office of Nuclear Regulatory Research (RES) carry out BWR stability analyses centered around fourteen specific questions. The questions originated from the demand to meet the General Design Criterion 12 in Appendix A of the Code of Federal Regulation 10CFR50, which requires that reactor power oscillations violating any Technical Specifications be either impossible by design, or detected and suppressed.

1.2 EPA Analyses of BWR Stability

One of the fourteen NRC-stipulated questions (see [Wulff et al. 1991], Section 1.2, for a complete listing) raised the issue of whether or not "available computer codes are reliable for predicting BWR instability". This question has been answered affirmatively for the BNL Engineering Plant Analyzer (EPA), on the basis of selected deterministic analyses (see Chapters 6, 7 and 9 in [Wulff et al. 1991]), with the primary focus on the EPA's ability to predict power and flow oscillations with large amplitudes. Over ninety transients of the BWR system have been simulated on the EPA [Wulff et al. 1991]. This number of simulations was performed (i) to resolve ten out of fourteen NRC-stipulated questions, (ii) to determine the effect of power oscillations on fuel and suppression-pool temperatures for ten NRC-selected ATWS scenarios (Anticipated Transient Without Scram), (iii) to rank modeling parameters according to their impact on stability, (iv) to check out results of BWR stability-related analyses submitted to the NRC by the industry, and (v) to assess the EPA's capability of analyzing BWR stability.

This report presents the statistical analysis of the uncertainty in simulating specifically the rise of suppression-pool temperature during an ATWS with power and flow oscillations, the specific scenario which was selected by the NRC to include some emergency response procedures (see Section 2.3 below). Thus, the work presented here supplements the previously performed EPA assessment (see [Wulff et al. 1991], Chapters 6, 7, and 9), as it quantifies the simulation uncertainty in a systematic and traceable way by applying the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology [Technical Program Group 1989, Boyack et al. 1990, Wilson et al. 1990, Wulff et al. 1990, Lellouche et al. 1990 and Zuber et al. 1990].

1.3 Need For Uncertainty Estimations

Safety analyses, carried out by computer simulation with complex systems codes, are generally not acceptable without a quantitative uncertainty or error estimation. Computer codes imply modeling assumptions; they contain correlations, many of which have been derived from data of small-scale test facilities. This introduces uncertainties of scale. Moreover, codes have so-called discretization errors in their procedures for numerically integrating the governing equations of mass, momentum and energy conservation, for the neutron kinetics, the heat transfer through the fuel elements, for the coolant, for all the rotating machinery and moving components, and for the dynamic equations describing the plant's control systems. Finally, the computer predictions are affected by the uncertainties introduced by the specification of boundary conditions. In fact, the uncertainty of deterministic computer predictions is dominated by nearly unpredictable (stochastic) operator interventions, the more the longer time that the predicted transient lasts.

The quantification of simulation uncertainties is therefore necessary.

Individual uncertainties have been assessed earlier [Wulff et al. 1991] by deriving analytical solutions, by nodalization studies, by performing separate computer studies, and by comparing simulation results with experimental data. These uncertainty estimates, however, did not produce the composite uncertainty.

1.4 CSAU as Basis for Uncertainty Evaluation

The NRC, its contractors and consultants have developed the previously mentioned CSAU evaluation methodology to quantify in a systematic way the overall uncertainty of scaling, of modeling and of computing, with a selected computer code, for a specifically selected power plant, a specifically selected accident scenario, and for the prediction of a particular, selected important safety parameter.

The CSAU evaluation methodology is the fundamental basis for the uncertainty estimation presented in this report. The scarcity of experimental data from Separate Effects Tests, and from Integral Effects Tests on transients with large-amplitude power and flow oscillations, however, limited the scope of the CSAU methodology application. Bounding calculations had to be used for bias estimations, which lead to results of greater uncertainty than could have been achieved with a larger data base.

1.5 Objectives of Uncertainty Analysis

It is the purpose of the uncertainty analysis presented in this report to determine the probability distribution function of the uncertainty associated with the EPA prediction of the maximum temperature reached in the suppression pool of the LaSalle-2 BWR power plant, when a Recirculation Pump Trip-induced ATWS, encountering large-amplitude power and flow oscillations, is being simulated. For this ATWS are specified a number of automatic and manual control actions, which are later described in Section 2.2.3, and intended primarily to protect the suppression pool from overheating.

The uncertainty estimation is to include all the relevant random and systematic (bias) uncertainties.

The suppression-pool temperature was selected by the NRC as being an important safety parameter for assessing containment integrity.

1.6 Organization of Report

Chapter 2 describes in Sections 2.1 and 2.2 the CSAU methodology as far as appropriate for the purpose of this report. The more detailed description is found in the previously cited references [Technical Program Group 1989, Boyack et al. 1990, Wilson et al. 1990, Wulff et al. 1990, Lellouche et al. 1990 and Zuber et al. 1990]. Section 2.3 describes then the selected reference ATWS transient, and Sections 2.4 through 2.8 the application of the CSAU methodology to the simulation of this transient on the EPA, starting with the ranking of relevant parameters, and the determinations of their respective uncertainty ranges.

The major steps in the CSAU methodology application are the determination of the EPA's applicability to the selected transient, presented in Section 2.5; the comparison of EPA predictions with plant data (Integral Effects Tests), as described in Section 2.6; the EPA simulations of the reference ATWS transient and of sixty-three transients with parameter variations within their uncertainty ranges. This is explained in Section 2.7.

Section 2.8 describes first how the response surface is then generated, as a surface fit to the EPA-predicted suppression-pool temperatures, which were obtained by simulating transients with the above-described variations of relevant modeling parameters.

Section 2.8 presents then the statistical analysis which leads to the probability distribution of the uncertainty in predicting the suppression-pool temperature.

Chapter 3 presents the conclusions.

2 CSAU Application to EPA Analyses

2.1 The Code Scalability, Applicability and Uncertainty (CSAU) Evaluation Methodology

The CSAU Evaluation Methodology facilitates in a practical and auditable way the systematic quantification of the uncertainty associated with the use of a particular computer code for the prediction of a particular systems transient in a particular power plant. The methodology has three elements [Boyack et al. 1990].

In the First Element, one establishes the computer code requirements from specifications of the transient of interest, and determines, on the basis of code documentation, whether or not the selected computer code can meet the requirements. To establish code requirements, one needs to identify and rank, in the order of their priorities, all processes in all system components. Then, with the help of code documentation, one needs to rank the modeling parameters in the order of their relevance to the predicted computer result. This ranking leads to the Phenomena Identification and Ranking Table (PIRT) [Wilson et al. 1990].

In the Second Element of CSAU [Wulff et al. 1990], one determines for each of the relevant modeling parameters the range of uncertainty, such as the 95% probability limits and the probability distribution. Where this is impossible, one uses extreme limits and uniform probability, as this provides the probability distribution of maximum ignorance and is most conservative. The probability distributions are estimated from scatter (providing, e.g., the standard deviation) of data from Separate Effects Tests (SET). Integral Effects Tests (IET) provide overall code uncertainties and are used to support the statistical uncertainty analysis. Systematic errors or biases are determined by comparing computer predictions with experimental results from IETs, and by performing separate hand calculations [Wulff 1988]. The computer and hand calculations are also needed to reduce the analysis efforts by eliminating from the PIRT listing all those parameters to which the computer results are not sensitive.

In the Third Element of CSAU, one determines the probability distribution function of the overall uncertainty for predicting the subject key (safety) parameter, such as the suppression-pool temperature. The 95% confidence

limits of uncertainty are then obtained from the distribution function [Lellouche et al. 1990].

The uncertainty probability distribution function is found by statistically evaluating the results of $\approx 10^4$ approximate systems simulations, carried out with modeling and operating parameters randomly varied within their respective uncertainty ranges. The large number of approximate system simulations are carried out by replacing the actual (often expensive) systems simulations by evaluating a response surface. This response surface is curve fitted to the results of a relatively small number (≈ 50) of system simulations: the first for the transient of interest, the others for transients with parameters varied randomly within the range of parameter uncertainty.

After shifting the probability distribution function to account for the systematic effects of biases, one obtains the uncertainty brackets at the 95-percentile points on the distribution function.

2.2 Scope of CSAU Application to BNL's EPA Analyses

The BNL Engineering Plant Analyzer (EPA) was used to simulate more than ninety transients related to the 1988 instability at the LaSalle-2 BWR power plant [Wulff et al. 1993]. Here, the purpose of the CSAU methodology application to the analyses performed on the EPA is to determine the uncertainty margin of predicting the maximum temperature reached in the suppression pool, for a particular, NRC-selected ATWS with power and flow oscillations.

All three elements of CSAU, as described above in Section 2.1, have been performed, except that IET data were not available for developing scatter diagrams of comparisons between EPA predictions and IET results.

2.3 Selection of Reference Transient

The Reference Transient with its set of defining parameters is the basis for the uncertainty analysis; all modeling uncertainties are variations about the defining parameter set. The Reference Transient is an ATWS. It is initiated by a Recirculation Pumps trip, which is followed by a Turbine trip, the failure of MSIV closure, and boron injection. In

view of the previously performed ATWS simulations [Wulff et al. 1993], the Reference Transient was started with the same LaSalle-2 plant conditions of March 9, 1988. The Reference Transient was selected by the NRC. The purpose is to determine the safety margin of exceeding the suppression-pool temperature limit, which is related to the containment integrity.

2.3.1 LaSalle-2 Plant Conditions Before 1988 Instability

On March 9, 1988 the LaSalle Unit 2 BWR plant was at steady-state conditions, operating at approximately 84% of rated power and 76% of rated core flow, with the feedwater temperature at 479 K (402 °F) and the control rods withdrawn to the 99% power line. The conditions at the plant prior to the event initiation are summarized in Table 2.1 below. The event sequence of the actual LaSalle-2 instability differs from that of the postulated ATWS and is therefore not relevant here; it is found in Section 2.2 of the BWR Stability Analysis report [Wulff et al. 1993].

2.3.2 Reference ATWS Transient

The transient was initiated from the above steady state by a trip of both Recirculation Pumps. The event sequence is summarized in Table 2.2 below.

The recirculation-pumps trip reduced the core flow rapidly to the flow rate of natural circulation, from 76% to 29% of normal full flow. The core flow reduction in turn decreased rapidly the fission power, from 84% to about 45% of full power, because of the increase in vapor void generation and void reactivity feedback. Because the fission power decreased, the dome pressure also decreased by 1.6 bar (24 psi), and the steam flow to the turbines, and therefore the extraction steam flow from the turbines to the feedwater heaters, decreased also, thereby providing increasingly insufficient feedwater preheating. The feedwater heater control system responded then to the loss of feedwater preheating and isolated automatically some of the steam extraction lines leading to the preheaters. This isolation is needed normally to prevent extraction steam condensate from backing up into the main turbine.

The scram signal was disabled at the outset to achieve the postulated ATWS.

The power oscillations grew and exceeded the scram setpoint of 118% at 6 min after the RCP trip. Nine minutes after RCP trip, the oscillation amplitude reached its maximum of 1,300%. Twelve minutes after RCP trip, the turbine was manually tripped, while allowing the Bypass Valves to open and disabling the closure of all the Main Steam Isolation Valves (MSIV). (No MSIV closure at low-level signal).

After two Safety Valves (SRV) had opened, the SRV Low-Low set-point logic was activated. Steam discharged from the vessel to the suppression pool. The Residual Heat Rejection System started up automatically at 12.64 min after RCP trip, when the pool temperature reached 311K (100°F). As the Suppression Pool temperature reached 317K, Boron injection was initiated, the Feedwater Pumps were tripped, and High Pressure Injection Cooling (HPCI) was inhibited in accordance with Emergency Operator Procedures (EOP).

As the coolant collapsed, the liquid level in the vessel reached the L2 setpoint at 13.79 min, Reactor Core Injection Cooling (RCIC) was turned on. The coolant reached the core with the delay of 25 s, and the electric motor-driven Feedwater Pumps were started up. At the time of 14.2min after the event initiating RCP trip, the coolant level control was started. The predicted collapsed liquid level L_c was fed into an operator simulating robot, who controlled reproducibly the feedwater and RCIC flows, so as to lower the coolant level to the elevation of $L_0 = 1.17$ m (46 in) above the top of active fuel, where the setpoint L_0 is selected to give a control margin against undershooting.

The robot controlled the fraction f of normal feedwater and RCIC flow rates, according to

$$\frac{df}{dt} = \frac{1}{\tau_{LVC}} (L_0 - L_c) / L_{\max} , \quad (2.1)$$

where $\tau_{LVC} = 20$ s is the time constant, selected to achieve the level drop in the NRC-specified time, and $L_{\max} = 10$ m is the level scale normally used in the EPA (which uses fractional arithmetic).

There were no other actions taken besides the event-initiating RCP trip, the turbine trip, the inhibition of scram, of MSIV closure and of HPCI, and the level control by robot. All other control actions were automatic.

**Table 2.1 LaSalle-2 Plant Conditions
Before Initiation of Instability**

Reactor Power	2,801	MWt	84% of full power
System Pressure	68.78	bar	(997.5 psia)
Core Inlet Flow	10,326	kg/s	(81.95 10 ⁶ lbm/hr)
			76% of full flow
Steam Flow Rate	1,415	kg/s	(11.23 10 ⁶ lbm/hr)
Feedwater Flow Rate	1,427	kg/s	(11.32 10 ⁶ lbm/hr)
Feedwater Temperature	478.9	K	(402.4 °F)

**Table 2.2 Event Sequence Summary
for Reference ATWS**

Time (min)	Action or Event
0.0	Steady state, 85% power, 75% core flow Event initiation: trip of both Recirculation Pumps, Scram disabled.
6.96	Feedwater preheating deteriorates, Power reached scram setpoint of 118% of full power,
9.0	Large amplitude power oscillations reached
12.0	Turbine tripped, MSIV closure inhibited
12.04	SRV Low-Low setpoint logic activated
12.64	Suppression Pool temperature reached 311K (100°F), RHR system starts up,
13.25	Suppression Pool temperature reached 317K (110°F) Boron injection activated with 60s delay, FW Pump tripped, HPCI inhibited per EOP, Control Rod Drive cooling flow rate at 0.96kg/s,
13.79	Vessel Water Level reached L2 setpoint, RCIC turned on with 25s delay, Electric FW Pump motors turned on,
14.20	Level Control started (FW control robot),
16.20	Suppression pool reaches peak of 329K (133°F),
18.50	Power dropped to decay heat level,
24.00	Simulation terminated.

At the time of 16.2 min after the RCP trip, the suppression-pool temperature reached its peak of 329 K (133°F). The reactor power fell to the decay-heat level at the time of 18.5 min after the RCP trip.

The simulation of the reference case was terminated after 24 min. This completes the description of the reference ATWS simulation.

2.4 The Phenomena Identification and Ranking Table (PIRT)

2.4.1 Purpose of PIRT

The Phenomena Identification and Ranking Table (PIRT) is developed to concentrate the uncertainty analysis efforts on the uncertainty propagation arising from only the relevant modeling parameters. Instead of attempting the impossible task of quantifying the effect of each and every parameter uncertainty on the overall uncertainty, one identifies and ranks in a table (PIRT) the manageable number of relevant phenomena taking place in each reactor component, and from that the manageable number of modeling parameters for which the uncertainty analysis must be carried out. It will be shown that only nineteen parameters impact the uncertainty analysis.

2.4.2 PIRT Application to Reference ATWS

The Phenomena Identification and Ranking Table (PIRT) is developed by a systematic evaluation of expert opinions concerning the identification and ranking of the phenomena which have the greatest impact on the suppression pool temperature rise during the postulated ATWS transient. In cases of disagreement between experts on the relevant phenomena ranking, such disagreements are resolved by analysis.

Members of the NRC-formed Technical Review Group (TRG) served as experts: Drs. H.S. Cheng, U.S. Rohatgi and W. Wulff from Brookhaven National Laboratory, Dr. G. Lellouche from S. Levy, Inc., Drs. G. Johnsen and S. Rouhani from the Idaho National Engineering Laboratory, and Dr. J. March-Leuba from Oak Ridge National Laboratory participated in developing PIRT. The tabulated ranking developed by the experts was evaluated at the Idaho National Engineering Laboratory and is reported in Sections 2.4.2.1 through 2.4.3 below.

2.4.2.1 Description of the PIRT Process

The PIRT development process consists of five major steps:

1. Identification of transient phases,
 2. Identification of system components,
 3. Identification of potentially important phenomena for each phase and in each component,
 4. Phenomena and component ranking, and
 5. Data processing.
1. The conceptual division of a transient into phases is necessary to break long and complicated transients into short time periods with simple processes; it is possible when the system behavior can be characterized by periods that are dominated by different phenomena, proceeding in distinct components at distinct time rates. The division into phases facilitates the ranking process since it precludes inappropriate pair-wise comparisons of disparate phenomena. It also ensures the retention of important phenomena that are not exhibited during all phases.
2. Phenomena or processes take place in system components. The reactor system is therefore divided into components, consistent with the individual component models in the computer code, but at the highest level possible so as to minimize the total number of components without compromising the ranking process. The level of system partitioning is governed by the need to account only for the global impact of the processes in the component on the safety parameter of interest; that is, for the present work on the suppression-pool peak temperature. Components that have little or no bearing on the safety parameter of interest are not included in the PIRT.
3. The identification of important phenomena (or processes, such as boiling, condensation, frictional dissipation of mechanical energy, valve actions, etc.) is the third and most important step in the PIRT development process. In this step the experts list for each of the identified system components that will materially affect the safety parameter of interest. A phenomenon may occur in only one component or in many.
4. Phenomena ranking is the ordering of the previously identified phenomena (thermohydraulic processes) according to their impact on the safety parameter of interest. Phenomena ranking leads also to the ranking of the components in which the phenomena take place. Components are ranked on the basis of the importance of all the

processes taking place in them during a phase. Component ranking is needed in Step 5 to check the consistency of the phenomenon ranking. Phenomenon ranking is the only objective and result of Step 4.

The ranking is carried out separately for each one of the phases identified in Step 1. Components are ranked pair-wise on the basis of how many important phenomena take place in them during a phase. Phenomena are then ranked pair-wise, based on the question: "how important to the safety parameter of interest is Phenomenon A in Component J, compared to phenomenon B in the same component J?"

In the present study, each component and each phenomena pair (A,B) are assigned one numerical value according to the ranking scale illustrated below:

<u>Score Value</u>	<u>Interpretation</u>
5	A is much more important than B,
3	A is moderately more important than B,
1	A is as important as B,
1/3	A is moderately less important than B,
1/5	A is much less important than B.

The values of 4, 2, $\frac{1}{2}$, and $\frac{1}{4}$ are reserved for compromise positions, and used when the group of experts seeks to reach consensus.

A set of component and phenomena tables is created, one for each of the phases identified in Step 1. The scores are entered in the component and phenomena tables in matrix form, each component and phenomenon appearing as a row and as a column. All diagonal entries are unity; only the matrix elements on the upper side of the diagonal are needed, since the elements on the lower side are reciprocals of the elements on the upper side. A convention is needed to associate Elements A in the table above with row labels.

5. The data are then processed by a computer program that executes the Analytical Hierarchy Process [Watkins 1992], to eliminate any inconsistency of the subjective ranking, and to quantify the consistency of the final component and phenomena ranking for each transient phase. The resulting phenomena ranking accounts for both the component and phenomena ranking data, because it combines both the component and phenomena ranking to compute the overall phenomena ranking. Thus if phenomenon A is relatively important for all compon-

ents, it ranks higher than a phenomenon B that is important in only one component.

2.4.2.2 Implementation of the PIRT Process

The postulated ATWS transient, described in Section 2.3.2, naturally divides itself into two phases. The first phase begins at time zero, when both Recirculation Pumps trip, initiating the transient. The first phase ends, and the second phase begins when the Low-Low SRV setpoint is reached.

During the first phase, the phase of power oscillation growth, there is no reactor coolant delivery to the suppression pool. Suppression-pool heating characterizes the second phase, which ends after the suppression-pool temperature has passed its maximum.

For the postulated reference transient, Phase 1 ends at 12.04 minutes, and Phase 2 at 24 minutes after the Recirculation-pump Trip (see Table 2.2).

A list of components was developed that contained sixteen components.

For Phase 1, some components were lumped together to simplify the ranking process. The turbine, condenser and condensate storage tank were combined into the steam and feedwater systems, constituting the balance of plant. The upper and lower plena, the separator, steam dryer and recirculation pumps were combined into the ex-core component group. The resulting eight components were ranked for Phase 1, first by expert judgement and then by the AHP computer code.

Component Ranking, Phase 1: Power Oscillation Growth

<u>Component</u>	<u>Rank</u>	<u>Score</u>
Core	1	8.2
Level Contr.	2	6.4
Oper/EOP Pressure Control	3	5.4
Steam/Feedwater	4	4.6
Ex-Core Comp.	5	4.0
ECCS	6	2.8
RHR	7	1.2
	8	1.0

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Oper/EOP represents a robot component which was implemented in the Engineering Plant Analyzer to simulate repeatedly operator actions for lowering the reactor vessel pressure and the mixture level elevation in the downcomer, as suggested by Emergency Operating Procedures. ECCS and RHR stand for Emergency Core Cooling System and Residual Heat Rejection, respectively.

Phase 2 is the more important one because its component list contains also the suppression pool, and the suppression-pool temperature reaches its peak during this phase. Fourteen components were ranked for Phase 2, without combining any components.

**Component Ranking,
Phase 2:
Suppression-Pool Heat-up**

<u>Component</u>	<u>Rank</u>	<u>Score</u>
Core	1	7.8
Level Control	2	7.0
Oper/EOP	3	6.8
Boron System	4	6.4
Feedwater System	5	6.2
Suppression Pool	6	5.6
Steamline and SRV	7	5.2
Pressure Control	8	3.6
Upper Plenum, Steam Separator	9	3.2
Lower Plenum, Downcomer	10	2.2
Autom. Depr. System (ADS)	10	2.2
ECCS	11	1.8
Condenser/ Condensate Storage Tank	11	1.8
RHR	12	1.6

The core, level control system, and operator/EOP (robot) are the top ranked components for both phases. This result reflects the perceived importance that the core power, the level control and operator actions have on suppression-pool temperature.

Eight thermohydraulic processes were identified as potentially important phenomena and ranked by the method described in the previous section. The phenomena and their ranking are presented below.

2.4.3 PIRT Results

The table below ranks the identified phenomena in the order of importance to the second, that is the more important phase. Shown are for every phenomenon its rank in the second and fourth columns, and its AHP score in the third and fifth columns.

The ranking differs from that of the original CSAU ranking [Wilson et al. 1990, Table 1, p. 23], in that it does not show the phenomenon ranking for each component. This shortcoming is compensated for in the process of selecting the relevant modeling parameters for each component model.

The lowest score in the ranking table below is 4.6, suggesting the need to include all phenomena in the uncertainty analysis.

<u>Phenomena</u>	<u>Phase 1</u>		<u>Phase 2</u>	
	<u>Rank</u>	<u>Score</u>	<u>Rank</u>	<u>Score</u>
Power				
Distrib.				
Dynamics	1	9.0	1	9.0
Two-Phase				
Flow				
Dynamics	4	6.2	2	6.8
Energy				
Transfer	3	6.6	3	6.6
Pressure				
Drop	5	6.0	3	6.4
Boron				
Mixing	-		4	5.6
Critical				
Flow	-		5	5.4
Phase-				
Change				
Processes	2	6.8	6	4.8
Downcomer				
Mixing	6	4.6	7	4.6

Power Distribution dynamics denotes core burn-up, axial and radial power distributions, and reactivity feedback mechanisms (void, moderator, Doppler). Their perceived importance on core power and energy deposition into the suppression pool is quite clear.

Two-Phase Flow Dynamics encompasses the interaction of liquid and vapor in the core, upper plenum, separators, and downcomer. In particular, it refers to phase separation which affects void profiles in the core and influences core power.

Energy Transfer refers to the heat transfer to the condensate in the preheaters, to heat transfer from the fuel to the coolant in the core, and to gamma heating of the reactor coolant. Feedwater preheating influences the subcooling of core inlet coolant, and thus core power. Core heat transfer and gamma heating directly influence the deposition of energy into the suppression pool. It is obvious that "energy transfer" involves four models for three components.

Pressure Drop refers to form loss and wall shear in the core. The overall pressure drop across the core affects the core flow, and thus the vapor void fraction and power. Form losses at the core inlet (single-phase liquid) stabilize, whereas core exit form losses (two-phase mixture) destabilize the core flow and power. The power amplitude affects the mean power and therefore the energy deposition into the suppression pool [Wulff et al. 1993, p. 48].

Boron Mixing in the reactor core affects the time it takes for the power to drop to the level of decay heat, and thereby the amount of heat deposition into the suppression pool.

Critical Flow at the safety and relief valves (SRV's) determines the time and duration of steam discharge

from the reactor pressure vessel to the suppression pool, and thereby the rate and duration of pool heat-up.

Phase-Change Processes are boiling and flashing in the core, flashing and condensing in the downcomer.

Downcomer Mixing refers primarily to the condensation of saturated steam onto the subcooled feedwater in the downcomer, once the coolant mixture level drops in the downcomer below the feedwater sparger. While carry-under steam and recirculation liquid always reach thermal equilibrium with the feedwater, the extent to which the subcooled feedwater reaches saturation before mixing with the recirculation flow depends on its exposure to steam along the elevation difference between sparger and mixture level. The preheating of the exposed feedwater by saturated steam determines the liquid subcooling at the core inlet, which in turn influences core voiding and power.

This completes the description of the ranked phenomena. Knowing the important phenomena to be modelled by the computer code (HIPA) of the Engineering Plant Analyzer, one must then determine from the code documentation whether the code is applicable to the simulation of the postulated ATWS (CSAU Element I, Step 6 [Boyack et al. 1990, p. 5], see also Section 2.1 above). If the code is applicable, then one must determine, also from the

Phenomena and Modeling Parameters

Power Distribution Dynamics

Reactivity feedback coefficients:

Void,
Doppler;
Radial peaking factor F_q ,
Axial peaking factor F_q ,
Boron worth.

Two-Phase Flow Dynamics Energy Transfer

Drift Flux Parameter C_o
Feedwater Heating heat transfer coefficient,
Fuel Gap Conductance,
Core heat transfer coefficient.

Pressure Drop

Core Outlet Form Loss Coefficient,
Core Inlet Form Loss Coefficient.

Critical Flow Phase Change Processes

SRV Flow Area.
Vapor generation rate (core)
Condensation rate (downcomer)

Downcomer Mixing

Condensation Rate,
Carry-under Fraction.

Other

Feedwater Controller Gain
Level Control setpoint
Level control time constant
SRV setpoints

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code documentation, the parameters which model the important phenomena (CSAU Element I, Step 5 [Boyack et al. 1990, p. 5]).

The code applicability is discussed in the next section. The relevant modeling parameters are listed below. All of the identified phenomena are represented by their controlling modelling parameters except boron mixing. Since the EPA models only axial (one-dimensional) boron distribution, but not lateral mixing, the modelling uncertainty for boron mixing is accounted for indirectly by varying boron worth (shown under "Power Distribution Dynamics").

The above list shows also, under the phenomena heading "other", the modeling parameters for the downcomer level, pressure and feedwater control systems, which are near the top of component ranking, but were not represented by their associated control processes in the phenomena identification.

The above key modeling parameters are listed with their reference values in Table 2.5. Their uncertainty brackets, as listed in Table 2.7, are based in part (for the reactor fuel, for example) on those used in the PWR CSAU study [Boyack et al. 1990], in part on separate effects tests (SRV flow uncertainty, private communication from G. Lellouche 1991), or on conservative estimates made by the Technical Program Group. The uncertainty brackets are deemed rather too large than too small.

Unknown probability distributions were taken to be uniform, representing maximum ignorance about the distribution, and producing the most conservative uncertainty propagation through the simulation.

2.5 EPA Applicability to the Reference ATWS

The High-Speed Interactive Plant Analyzer code HIPA in the Engineering Plant Analyzer (EPA) has suitable models for the dynamics of nonhomogeneous, nonequilibrium two-phase flow, based on the four-equation drift-flux model [Wulff et al. 1984]. HIPA has appropriate models for simulating transient fission power, fuel conduction, nonequilibrium boiling heat transfer, vapor generation and condensation, form loss, and wall friction.

HIPA models the steam line flow, including acoustic effects, the turbines with steam extraction for driving the feedwater pump turbine and for feedwater preheating, the

condenser, the feedwater preheaters, the dynamics of the feedwater pump, and of the recirculation loops.

HIPA simulates dynamically all reactor control systems for coolant level and feedwater control, for pressure and for power controls. It simulates all engineered safety systems.

The EPA has been used successfully for ATWS simulations [Wulff et al. 1991]. HIPA has been shown to be reliable for predicting and analyzing BWR instability. This was demonstrated [Wulff et al. 1991] by comparing EPA predictions with plant data, with results from the frequency-domain code LAPUR, and with results from analytical solutions and independent numerical analyses.

2.6 EPA Assessment with Integral Effects Test Results

The BNL Engineering Plant Analyzer (EPA) has been assessed in two ways for stability-related analyses with integral facilities:

(a) The EPA has been assessed with plant data from three Peach Bottom Stability Tests, covering **small-amplitude** oscillatory transients [Wulff et al. 1991, Sections 7.3 and 7.4].

The comparison of the EPA predictions with plant data from the Peach Bottom Stability Tests showed that the EPA is reliable for analyzing small-amplitude oscillations, involving neutron kinetics, thermal fuel response, coolant thermohydraulics, and control systems. The EPA is a time-domain code, but was assessed on the basis of frequency-domain criteria.

The EPA predicted for Peach Bottom Tests PT1, PT2, and PT4 the power to pressure transfer function gain with the biases and standard deviations of $-10 \pm 28\%$, $-1 \pm 40\%$, and $+28 \pm 52\%$, respectively.

The respective frequencies at peak gain were predicted with errors of $+6\%$, $+3\%$, and -28% .

The differences between experiments and EPA predictions are comparable to the associated experimental uncertainties ([Wulff et al. 1991], Figs. 7.2, 7.3, and 7.4); they are larger than the discrepancies between RAMONA-3B results and data from non-nuclear FRIGG experiments ([Rohatgi et al. 1990, 1991], Sect. 7.2, p. 118), and they are all smaller than the differences between

non-nuclear test data and results from the frequency-domain code NUFREG, as reported by Yadigaroglu ([Delhay et al. 1981], p. 376), and from the time-domain code TRAC-BF1 ([Rouhani 1990], Sect. 7.2, p. 119).

(b) The EPA has been assessed also with data obtained from the STARTup Transient REcording (STARTREC) System of the LaSalle-2 Power Plant during the March 9, 1988 instability event ([Wulff et al. 1991], Section 2.5). No other plant data exist from large-amplitude oscillatory transients.

The comparisons of EPA results with STARTREC data show that the EPA can be used for analyzing reliably core-wide, in-phase, thermohydraulic instabilities as had occurred at LaSalle-2.

Seven parameters are available from the STARTREC plant recording system and were compared with EPA predictions: fission power, reactor vessel pressure, core inlet flow rate, coolant level in the downcomer, feedwater temperature, and the flow rates of steam and feedwater.

The timing of the power oscillations, i.e., the onset of oscillations and the occurrence of the scram trip are predicted correctly.

The EPA-predicted growth of the power amplitude prior to reactor scram could not be compared with plant data, because the strip chart recordings did not cover a sufficiently long time span of the transient.

The EPA-predicted frequency of the power and flow oscillations is 11% smaller than the recorded frequency, and the predicted time-mean fission power prior to scram is 4.7% larger than the (presumably time-averaged) recorded fission power, with the standard deviation of 4.2%.

The EPA-predicted reactor pressure agrees with the STARTREC recording within 2.4% of the recorded value.

The predicted core inlet flow agrees with the plant data within plotting accuracy.

The EPA-predicted collapsed-liquid level differs from the recorded level by 7 cm (2.8 in), with the standard deviation of 9 cm (3.5 in).

The EPA-predicted feedwater temperature agrees within plotting accuracy with the recorded feedwater temperature.

The EPA overpredicted the steam and feedwater flow rates by $36 \pm 8\%$ and 22%, respectively, because the feedwater regulator failure, which had actually occurred, could not be simulated due to a lack of information. It was shown ([Wulff et al. 1991] p. 27), that the recorded feedwater is inconsistent with the recorded steam flow and coolant level; the feedwater flow recording is suspect.

The differences between plant data and EPA-predicted steam and feedwater flow conditions reflect the uncertainty of available information about the plant conditions just prior to a component failure, and during an instability.

Finally, the EPA-predicted relation between fission power and core flow was also assessed with data provided by the vendor. It was shown ([Wulff et al. 1991] p. 111) that the EPA predicts the core flow along the lines of constant control-rod position (CRL) in the power versus core flow map as follows:

Along the line of natural circulation, the EPA underpredicts the core flow rate by the mean of 1.2% and the standard deviation of 4.0%, with all the underpredictions above 10% of rated power.

Along the 100% CRL, the differences between the Power vs. Flow Map according to GE, and the EPA predicted thermal power for a given flow rate is 1.3% at 28% of rated flow; it is 2.4% at 75% of rated flow. Approximately the same result is obtained for the EPA-predicted 80% Control Rod Line (CRL).

All the differences between the Power versus Flow Map actually used by the utility, and the EPA predictions are within the uncertainty range of the GE-generated map.

2.7 ATWS Simulations with EPA

Eleven ATWS scenarios were simulated as part of the BWR stability analysis ([Wulff et al. 1991] Chapter 4). Because of the significance of ATWS and its potential impact on containment heating, the NRC selected the ATWS described in Subsection 2.3.2 above as the basis for the uncertainty analysis.

2.7.1 Reference ATWS Simulation

The initial conditions and the sequence of events for the NRC-selected reference ATWS are presented above in Section 2.3 and summarized in Tables 2.1 and 2.2.

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The objective is to determine the maximum temperature $(T_p)_{\max}$ reached in the suppression pool.

The Safety and Relief Valve setpoints are important for the prediction of the steam discharged into, and the heat-up of the suppression pool. The low-low setpoints were used after the first valve opening; they are listed in Table 2.3 below. The reactivity feedback coefficients for the reference ATWS are listed in Table 2.4 and defined by the following polynomials:

(i) for the void coefficient in the j-th computational cell:

$$C_{v,j} = C_{v,1} + C_{v,2}\alpha_j + C_{v,3}\alpha_j^2 \quad (2.2)$$

where α_j is the void fraction in cell j;

(ii) for the Doppler coefficient in the j-th computational cell:

$$C_{d,j} = (C_{d,1} + C_{d,2}T_{f,j} + C_{d,3}T_{f,j}^2) \cdot (1 + C_{d,4}\alpha_j + C_{d,5}\alpha_j^2), \quad (2.3)$$

where $T_{f,j}$ is the fuel temperature in the j-th cell; and

(iii) for the moderator temperature coefficient:

$$C_{m,j} = C_{m,1} + C_{m,2}T_{m,j} + C_{m,3}T_{m,j}^2, \quad (2.4)$$

where $T_{m,j}$ is the moderator temperature in cell j.

Table 2.3 Safety and Relief Valve* Setpoints for Reference ATWS Simulation

	<u>Bank 1</u>	<u>Bank 2</u>	<u>Bank 3</u>	<u>Bank 4</u>
<u>Normal operation</u>				
No. of SRVs	6	4	8	18
Flow Area Per Bank (cm ²)	2704.3	556.7	566.0	1393.0
Opening Pressure (MPa)	7.5207	7.6586	7.7966	8.4104
(psig)	1076	1096	1116	1205
Closing Pressure (MPa)	6.8310	6.9000	7.0379	7.7900
(psig)	976	986	1006	1145
<u>Low-low Set Point Operation</u>				
No. of SRVs	1	1	4	1
Flow Area Per Bank (cm ²)	135.2	135.2	277.2	069.9
Opening Pressure (MPa)	7.0379	7.3138	7.5897	7.6586
(psig)	1006	1046	1086	1096
Closing Pressure (MPa)	6.2793	6.4862	6.6241	6.6241
(psig)	896	926	946	946

*18 Relief Valves in Banks 1-3
18 Safety Valves in Bank 4

Table 2.4 Reactivity Feedback Coefficients for Reference ATWS Simulation

Parameter		Value
Void Coefficients	$C_{v,1}$	Proprietary GE Data, Peason 1988, NEDE-31428P
	$C_{v,2}$	
	$C_{v,3}$	
Doppler Coefficients	$C_{d,1}$	-2.5410E-05
	$C_{d,2}$	1.2607E-08
	$C_{d,3}$	-2.2978E-12
	$C_{d,4}$	0.3889
	$C_{d,5}$	0.2778
Moderator Temperature Coefficients	$C_{m,1}$	-2.5390E-05
	$C_{m,2}$	8.3430E-07
	$C_{m,3}$	-5.6144E-09

Figures 2.1 through 2.7 present the results of the reference ATWS simulation.

Figure 2.1 shows the Suppression Pool temperature and steam flow through the Safety and Relief Valves (SRV). Figure 2.2 shows the fission power and vessel pressure, Figure 2.3 core inlet flow and collapsed liquid level. Figure 2.4 presents the mass flow through the steam lines and the feedwater flow, Figure 2.5 the feedwater temperature and the void reactivity. Figure 2.6 shows the core inlet subcooling temperature and the Doppler reactivity. Finally, Figure 2.7 presents the boron concentration in the core and Reactor Core Injection Cooling RCIC flow.

2.7.2 Parameter Variations

Table 2.5 below presents the nineteen relevant parameters, along with appropriate reference values, as determined from PIRT, the Process Identification and Ranking Table of Section 2.4.3. These parameters are needed to subnd the response surface for the uncertainty analysis.

Table 2.6 below defines the initial axial power shape through twelve power peaking factors, one for each computational cell along the core. Table 2.6 gives also the 2σ uncertainties of the local power peaking factors. The axial power shape was distorted, the mean of the peaking factors was kept at unity.

The nineteen relevant parameters listed in Table 2.5 below were varied within their range of uncertainties, one at a time for each simulation. The following four subsections describe the variation of feedwater heating, convective heat transfer coefficients, condensation in the downcomer, and the drift velocity.

2.7.2.1 Modeling Uncertainty of Feedwater Heating

The feedwater preheaters are modeled in the EPA as the combination of a drain cooler and the main heat exchanger. The condensate at the main section's shell side is at saturation temperature, the drain cooler is under counter current single-phase flow conditions.

Table 2.5 Summary of Reference Parameter Set for Uncertainty Analysis

Parameter	Value
1 Outlet Form Loss Coefficients	
Avg. Core Channel	1.34
Hot Channel	1.34
Bypass Channel	8.0
2 Inlet Form Loss Coefficients	
Avg. Core Channel	27.0
Hot Channel	27.0
Bypass Channel	230.0
3 Void Coefficients, see Table 2.4	
4 Doppler Coefficients, see Table 2.4	
5 Radial Peaking Fct.	1.43
6 Axial Peaking Fcts., see Table 2.6	
7 Feedwater Heating, see Sect. 2.7.2.1	
8 Boron Worth (\$)	-58.61
9 Fuel Gap Conductance ($W/(m^2C^\circ)$)	5678.0
10 Heat Transfer Coefficients, see Section 2.7.2.2	
11 Condensation in Downcomer, see Section 2.7.2.3	
12 Drift Flux see Section 2.7.2.4	
13 Vapor Generation Rate, see Sect. 2.7.2.5	
14 Feedwater Controller Gain	1.0
15 SRV Pressure Setpoints see Table 2.3	
16 SRV Flow Area, see Table 2.3	
17 Level Control Time Constant (s)	20
18 Level Control Setpoint above TAF (m)	1.17
19 Carry-under Fraction	0.0

Table 2.6 Power Peaking Factors for Reference Axial Power Shape and 2σ Variations of Bottom Peaking Uncertainty ($\pm 6\%$).

Cell j	Reference	Perturbances $+2\sigma$	-2σ
Bottom of Core			
1	1.1400	1.1720	1.1078
2	1.2341	1.2688	1.1993
3	1.2239	1.2583	1.1894
4	1.1646	1.1973	1.1318
5	1.0851	1.1156	1.0545
6	1.0408	1.0389	1.0427
7	1.0159	0.9836	1.0483
8	1.0184	0.9861	1.0509
9	1.0206	0.9882	1.0531
10	0.9316	0.9020	0.9613
11	0.6936	0.6716	0.7157
12	0.4314	0.4176	0.4452
Top of Core			

As shown in [Wulff et al. 1984], the feedwater temperature at the preheater exit is

$$T_{FW,e} = T_{FW,i} + f_{FW}(T_{XTR} - T_{FW,i}), \quad (2.5)$$

where

$$f_{FW} = 1 - \frac{e^{m-n}(q-1)}{qe^{m-1}}$$

$$m = \frac{(AU)_{DRC}}{(c_p)_1 W_{FW}} \left[\frac{W_{FW}}{W_{XTR}} - 1 \right]$$

$$n = \frac{(AU)_{MC}}{(c_p)_1 W_{FW}}$$

$$q = W_{FW}/W_{DRC}$$

Here $T_{FW,i}$ and T_{XTR} are the feedwater and extraction steam inlet temperatures, respectively; A and U are the area and overall coefficient of heat transfer, respectively; $(c_p)_1$ is the specific heat of feedwater, and W stands for the mass flow rate; the subscripts DRC, MC, XTR, FW, i and e denote, respectively, drain cooler, main section, extraction steam, preheater inlet and exit.

Figure 2.1 Reference ATWS Scenario (Sect. 2.3.2)
Suppression Pool Temperature and SRV Mass Flow Rate.

Figure 2.2 Reference ATWS Scenario (Sect. 2.3.2)
System Pressure and Reactor Power.

Figure 2.3 Reference ATWS Scenario (Sect. 2.3.2) Collapsed Liquid Level and Core Inlet Flow Rate. (Level reference elevation is at downcomer entrance, or 201.42 in above top of active fuel).

Figure 2.4 Reference ATWS Scenario (Sect. 2.3.2) Steamline Entrance and Feedwater Mass Flow Rates.

**Figure 2.5 Reference ATWS Scenario (Sect. 2.3.2)
Feedwater Temperature and Void Reactivity.**

**Figure 2.6 Reference ATWS Scenario (Sect. 2.3.2) Core Inlet Subcooling
Temperature and Doppler Reactivity.**

Figure 2.7 Reference ATWS Scenario (Sect. 2.3.2) Core-averaged Boron Concentration and RCIC Mass Flow Rate.

The uncertainty of predicting the heat transfer coefficient (AU) is accounted for in the uncertainty analysis by changing m and n each in the total uncertainty bracket of $\pm 30\%$. At normal operating conditions

$$\begin{aligned} m &= 1.290 \\ n &= 2.223 \\ q_0 &= 2.876. \end{aligned}$$

The mass flow ratio q is computed from the transient mass flows.

2.7.2.2 Modeling Uncertainty of Convective Heat Transfer

The single-phase heat transfer coefficient is computed according to Dittus-Boelter, from Reynolds (N_{Re}) and Prandtl (N_{Pr}) numbers, from the thermal conductivity k_l and the hydraulic diameter d_h for heating:

$$h_{2-\phi} = C_{JL} e^{(6.46 \cdot 10^{-7} m^2/M) P} [T_{wall} - T_{sat}]^3$$

(2.7)

and the coefficient $C_{JL} = 2.555 \text{ W}/(\text{m}^2 \text{K}_4)$; it is varied in its total uncertainty bracket from -25% to $+50\%$ of its value.

2.7.2.3 Modeling Uncertainty of Condensation in Downcomer

The volumetric rates of vaporgeneration and condensation in the downcomer portion, where direct contact is possible between the incoming subcooled feedwater and saturated steam, are modeled in the EPA as the superposition of flashing, contact condensation, and mixing effects:

$$\Gamma_v = [(1-f_\psi) \Gamma_v^{pr} + f_{ct} \Gamma_v^{ct} + f_\psi \Gamma_v^{blk}]_k, \quad (2.8)$$

where

$$\Gamma_v^{pr} = \frac{1}{h_{fg}} [1 - (1-\alpha) f_\psi h'_f - \alpha h'_g] \frac{dp}{dt}$$

$$\Gamma_v^{ct} = \frac{W_{FW}}{V_0 h_{fg}} (h_{FW} - h_f)$$

$$f_{ct,k} = \left[\frac{(k-1) - f_\psi}{3} \right]^{1/2}, \quad k = 2, \dots, 4$$

$$(h_c)_{1-\phi} = C_{DB} N_{Re}^{0.8} N_{Pr}^{0.4} \frac{k_l}{d_h} \quad (2.6)$$

The coefficient $C_{DB} = 0.023$ and has the total uncertainty range from -25% to $+50\%$ of its value. The range is skewed because $C_{DB} = 0.023$ of the Dittus-Boelter correlation is for internal tube flow and underpredicts the heat transfer in rod bundle flow.

The Jens-Lottes correlation is used for boiling heat transfer

Here the first two terms on the right-hand side account for phase change above the mixture level, in the fraction $(1-f_\psi)$ of the k^{th} computational cell, while the third term accounts for the phase change below the level. Γ_v^{pr} denotes the rate of equilibrium flashing or condensing which is caused by pressure variations. Γ_v^{mx} is the rate of complete contact condensation of vapor on subcooled liquid, taking place while the liquid is dropping through the volume V_0 below the feedwater sparger. Γ_v^{blk} is the standard phase change rate for the mixture below the froth level, in the fraction f_ψ of the computational cell. The fraction f_ψ is computed from the mixture elevation,

which is obtained by integrating the level velocity derived from the phasic mass jump conditions.

The fraction f_{ct} determines the extent to which the feedwater has reached saturation before it hits the mixture level; the fraction starts out with zero at the sparger elevation where $f_v=1$ and $k=2$, and it relaxes asymptotically to $f_{ct}=1$ at 4.5m below the sparger, where $f_v=0$ and $k=4$. The exponent $1/2$ is taken from experiments and determines the rate at which saturation is approached.

The contact condensation uncertainty of $\pm 25\%$ is taken into account by multiplying the second term on the right-hand side by (1 ± 0.25) .

2.7.2.4 Drift Flux Modeling Uncertainties

The area-averaged vapor volumetric flux j_v is computed in the EPA in terms of the distribution parameter C_0 and the void-weighted, area averaged drift velocity $\langle v_{gj} \rangle$ as follows:

$$j_v = \alpha [C_0 j_m + \langle v_{gj} \rangle] \quad (2.9)$$

Where α and j_m denote, respectively, the area-averaged vapor void fraction and mixture volumetric flux. Both C_0 and $\langle v_{gj} \rangle$ are computed from correlations by Ishii [Ishii 1977, Eqs.(71) and (76)], except that the coefficient of $\langle v_{gj} \rangle$ was changed from $\sqrt{2}$ to 2.9 [Wulff et al. 1991, p. 20], to simulate external tube bundle, rather than Ishii's internal tube, flow conditions.

The C_0 effect overwhelms the $\langle v_{gj} \rangle$ effect on j_v . Therefore, the entire uncertainty of predicting j_v is accounted for by perturbing C_0 according to

$$\begin{aligned} j_v &= \alpha [C_0^* j_m + \langle v_{gj} \rangle] \quad (2.10) \\ C_0^* &= C_0 (1 \pm \epsilon_{C_0}) \mp \alpha \epsilon_{C_0} \\ \epsilon_{C_0} &= 0.17. \end{aligned}$$

Equation (2.10) was chosen to satisfy the physical requirements, that $C_0^*=0$ for $\alpha=0$ and $C_0^*=1$ for $\alpha=1$. The bracket of $\pm 17\%$ for ϵ_{C_0} covers the $\pm 2\sigma$ uncertainty range of Ishii's data.

This completes the discussion of the parameter uncertainty ranges for those parameters which were identified in Section 2.4.3 as important for inclusion in the uncertainty analysis.

2.7.3 Data Base for Response Surface

Section 2.7.1 above presents the reference ATWS simulation. Table 2.7 below presents the results obtained from repeating the ATWS simulation, with the nineteen parameters listed in Column 2 varied, one at a time, by the amounts shown in Column 3.

The reference values of these parameters are listed above in Table 2.5, if only a single value is needed. Table 2.3 shows the parameter values for SRV pressure set points and the SRV valve flow areas, and in Tables 2.4 and 2.6 for the kinetic parameters. The remaining parameters are first defined and then specified in Sections 2.7.2.1, 2, 3 and 4.

The third column in Table 2.7 shows the parameter variation, and the fourth column lists the EPA-simulated maximum Suppression Pool Temperature obtained with these parameter variations. The third and fourth columns define the response surface; they list the data to which the response surface is fitted.

The fifth column lists the time it took the Suppression Pool Temperature to reach its maximum shown in Column 4, the time being counted from the time $t=12$ min of the turbine trip.

The total of 63 simulations was performed to subtend the response surface. Section 2.8 presents the curve fitted response surface and the statistical uncertainty analysis.

2.7.4 Response Surface Topography

The response-surface data were computed for the NRC-specified BWR ATWS as described in Section 2.3. The simulation results showed some apparent anomalies. These anomalies are described and explained below.

Table 2.7 Response-Surface Data for Assessing Modeling Uncertainty of BWR Suppression-Pool Temperature Predictions: Recirculation-Pump-Trip-Induced ATWS with Power Oscillation

Simul. No.	Parameter	Variation of Ref. Val.	Max. Pool Temperature T_p (°F)	Time to Max.Temp. t_{Tp} (min)
1	1 Outlet Loss Coefficients	+ 30%	135.71	4.63
2		+ 15%	135.36	4.62
3		+ 5%	134.87	4.61
4		- 5%	132.34	4.05
5		-15%	127.05	3.03
6		-30%	126.95	3.00
7	2 Inlet Loss Coefficients	+ 30%	125.65	3.12
8		+ 15%	127.50	3.41
9		-15%	135.40	4.50
10		-30%	135.96	4.52
11	3 Void Coefficients	+ 20%	128.80	3.32
12		+ 15%	131.17	3.91
13		-15%	133.50	4.79
14		-20%	134.35	5.00
15	4 Doppler Coefficients	+ 15%	127.23	3.30
16		-15%	135.93	4.52
17	5 Radial F_q	+ 6%	134.30	4.62
18		-6%	128.29	3.12
19	6 Axial F_q	+ 6%	126.95	2.90
20		-6%	134.88	4.62
21	7 FW Heating	+ 30%	130.90	3.83
22		-30%	135.14	4.60
23	8 Boron Worth	-5%	133.54	4.34
24	9 Fuel Gap Conductance	+ 900%	133.52	4.46
25		-80%	132.23	5.20
Ref. ATWS		0	133.02	4.20

Table 2.7 (cont.) Response-Surface Data for Assessing Modeling Uncertainty of BWR Suppression-Pool Temperature Predictions: Recirculation-Pump-Trip-Induced ATWS with Power Oscillation

Simul. No.	Parameter	Variation of Ref. Val.	Max. Pool Temperature T_p (°F)	Time to Max. Temp. t_{Tp} (min)
26	10 Heat Transfer Coefficient	+ 50%	135.20	4.60
27		-25%	129.10	3.32
28	11 Condensation in Downcomer	+ 25%	133.30	4.26
29		-25%	133.28	4.30
30	12 Drift Flux	+ 17%	128.67	4.90
31		-17%	134.50	4.30
32	13 Vapor Generation	+ 20%	126.18	3.31
33		-20%	134.32	4.63
34	14 Feedwater Controller Gain	+ 25%	135.13	4.63
35		-25%	127.50	3.19
36	15 SRV Pressure Setpoints	+ 5%	124.10	2.73
37		-2%	157.53	7.80
38		-5%	170.80	12.23
39	16 SRV Flow Area	+ 10.0%	126.93	2.85
40		+ 7.5%	126.38	2.87
41		+ 5.0%	126.95	2.93
42		+ 4.0%	133.30	4.20
43		+ 3.0%	135.61	4.58
44		+ 2.0%	135.26	4.70
45		- 0.0625%	133.00	4.2
46		- 0.125%	135.13	4.5
47		- 0.25%	128.52	3.1
48		- 0.5%	126.95	2.6
49		- 1.0%	126.90	2.7
50		- 2.0%	129.08	3.35
51		- 3.0%	132.47	3.28
52		- 4.0%	126.56	2.90
53		- 4.5%	130.88	3.90
54		- 5.0%	134.55	4.58
55		- 7.5%	134.64	4.63
56		-10.0%	134.78	4.70
Ref. ATWS		0	133.02	4.20

Table 2.7 (cont.) Response-Surface Data for Assessing Modeling Uncertainty of BWR Suppression-Pool Temperature Predictions: Recirculation-Pump-Trip-Induced ATWS with Power Oscillation

Simul. No.	Parameter	Variation of Ref. Val.	Max. Pool Temperature T_p (°F)	Time to Max. Temp. t_{Tp} (min)
57	17 Level Control Time Constant	+25%	132.73	4.17
58		-25%	133.80	4.39
59	18 Level Control Setpoint	+20%	131.45	3.90
60		+10%	132.25	4.00
61		-10%	134.58	4.59
62	19 Carry-under Fraction	0.001	131.96	4.01
63		0.002	128.00	3.05
Ref. ATWS		0	133.02	4.20

2.7.4.1 Types of Anomalies

Two types of apparent anomalies were observed:

(a) Asymmetry:

Figures 2.8 to 12 show the trends of pool temperature variation with respect to the nineteen parameters listed in Table 2.5. As any one of the seven parameters shown in Figures 2.8 and 2.9 was changed, one at a time, from its reference value and within its uncertainty range, the predicted suppression-pool temperature varied asymmetricaly about its reference value. This is shown in Figures 2.8 and 2.9: positive and negative parameter changes produced for the six parameters shown in Figure 2.8 consistently smaller increases than decreases; the opposite was observed for the variation of Safety and Relief Valve (SRV) pressure setpoints, shown in Figure 2.9.

(b) Extrema In The Variation Of Suppression-Pool Temperature With Safety And Relief Valve Area:

The $\pm 10\%$ variation of Safety and Relief Valve (SRV) area, which accounts for the total uncertainty of predicting the flow rate of choked steam discharge from these

SRVs into the Suppression Pool, produced two unexpected extrema in the Suppression Pool temperature, as shown in Figure 2.10. The energy discharged from the reactor vessel into the Suppression Pool was expected to vary monotonically with the SRV flow rate.

2.7.4.2 Summary of Findings

(a) The predicted trends of Suppression Pool Temperature were found to be real and consistent with the ongoing processes.

(b) Two additional extrema have been found in the dependence of the maximum Suppression Pool Temperature on Safety and Relief Valves (SRV) flow rate.

(c) Hysteresis effects from opening and closing of SRVs can cause discontinuities in the dependence of peak pool temperature on SRV flow rate.

Simulated histories of pressure, fission power, core flows, steam line, SRV, and feedwater flows, coolant levels, boron concentrations, and reactivities have been consulted, and the above conclusions were reached as follows.

Figure 2.9 Variation of Maximum Suppression Pool Temperature With Variations of SRV Set Points. Low-low set point mode is in effect. All valve bank set points are changed as indicated, by the same amount.

Figure 2.8 Variation of Maximum Suppression Pool Temperature as a function of relative variations of parameters listed on the right side.

Figure 2.10 Variation of Maximum Suppression Pool Temperature with Variations of SRV Flow Area. All valve bank areas are changed by the same fraction as indicated on the x-axis.

Figure 2.11 Variation of Maximum Suppression Pool Temperature as functions of relative variations of parameters listed on the right side.

Figure 2.12 Variation of Maximum Suppression Pool Temperature as functions of relative variations of parameters listed on the right-hand side.
 τ time constant; G gain.

2.7.4.3 Explanations of Anomalies

(a) Bases: The maximum Suppression Pool temperature rise is

$$(\Delta T)_{\max} = \frac{\int_{t_i}^{t_o} (h_{\text{SRV}} - h_i) W_{\text{SRV}} dt}{c_p (M_i + \int_{t_i}^{t_o} W_{\text{SRV}} dt)}, \quad (2.11)$$

where h_i and M_i are the fixed initial enthalpy and mass of the water in the suppression pool, while h_{SRV} and W_{SRV} are, respectively, the total enthalpy and the mass flow rate of the steam entering the Suppression Pool through the Safety and Relief Valves (SRV). The first valve bank opens at the time t_i , the same valve bank closes as the last bank at time t_o .

The number of SRVs that discharge steam, the duration of steam discharge, the total enthalpy h_{SRV} and the steam mass flow rate W_{SRV} all depend upon the variation in time of the reactor vessel pressure p and on the pressure setpoints at which the valves open and close. The setpoints for valve opening differ from the setpoints for valve closing. This difference introduces very important hysteresis effects.

The reactor vessel pressure history is governed by mass and energy balances, in conjunction with the thermal and caloric equations of state. These determine the rate of pressure change. For the purpose of understanding the anomalies listed under 2.1 and 2.2 above, it is sufficient to approximate the rate of change of pressure here by the equation for thermal equilibrium (the general equation is found on Page 3-43 of NUREG/CR-3943, "The BWR Plant Analyzer" (1984):

$$\frac{dp}{dt} = \frac{Q \frac{v_{fg}}{h_{fg}} + \frac{W_{FW}}{\rho_f} - \frac{W_{SL}}{\rho_g}}{V \left[\frac{v_{fg}}{h_{fg}} (1 - \rho_f h'_f) + \frac{\rho'_f}{\rho_f} + \alpha \left[\frac{v_{fg}}{h_{fg}} (\rho_g h'_g - \rho_f h'_f) + \frac{\rho'_g}{\rho_g} - \frac{\rho'_f}{\rho_f} \right] \right]}, \quad (2.12)$$

Thus, the maximum Suppression Pool temperature rise depends on the vessel pressure-dependent steam discharge rate W_{SRV} and steam total enthalpy h_{SRV} , and on the duration of discharge, more specifically on the valve closing time t_o .

The maximum Suppression Pool temperature rise is therefore governed by

- (i) the number of SRVs that open, and
- (ii) the length of time that each bank of SRVs remains open. The variation of pool temperature rise with respect to variations of SRV area (Type 2 anomaly) is governed strongly by the time t_o at which the last Relief Valve Bank closes (the bank that opens first and closes last at the lowest pressure set point for valve reseating).

All the pressure-dependent properties of saturated liquid and vapor (subscripts f and g, respectively), namely the densities ρ , the enthalpies h , the specific volume $v_g = 1/\rho_g$, $1/\rho_f$ and enthalpy h_{fg} of phase change, as well as the derivatives with respect to pressure (denoted by superscripted primes) depend all on pressure and vary for all cases according to the same narrow pressure range between pressure setpoints. Thus, the depressurization rate varies from case to case as the heating power Q , the steam discharge rate $W_{\text{SL}} = W_{\text{SRV}} + W_{\text{BYP}}$, the feedwater injection rate W_{FW} , and the void fraction α in the reactor vessel vary from case to case. W_{BYP} is the Turbine Bypass flow rate.

(b) Explanation of Asymmetry:

Asymmetries are the result of two causes. The first cause is the system's nonlinearity. The second cause arises from the need of changing the modeling parameter of interest, while starting for all cases from the same initial reactor

power and flow conditions, which is achieved by changing, in addition to the design parameter of interest, also the control-rod positions.

(c) Nonlinearity. The reactor system response to parameter variations is extremely nonlinear under the reference conditions, because the system is unstable, yet close enough to the stability border that the change in one direction, of one in a class of modeling parameters, renders the reactor stable, while the same change in the opposite direction of the same parameter renders the reactor only slightly more unstable.

A typical example is the variation of the drift-flux parameter C_o for the relative vapor velocity in the reactor core. Figure 2.13 shows the System Pressure and the Reactor Power for three C_o values. Figure 2.13a at the top is for $C_o = 1.17(C_o)_{REF}$, Figure 2.13b in the middle is for the reference case with $C_o = (C_o)_{REF}$, while Figure 2.13c at the bottom shows the result for $C_o = 0.83(C_o)_{REF}$. The largest C_o -value renders the reactor stable; the reference case and the case with the smallest C_o -value show large power and pressure oscillations. As a result, the peak pool temperature of 326.86 K (128.67°F), obtained with the largest C_o -value, differs by the much larger amount of 2.42 K (4.35°F) from the reference peak pool temperature of 329.27 K (133.02°F), than the 0.82 K (1.48°F) difference of the 329.84 K (134.05°F) peak pool temperature obtained with the smallest C_o -value. That explains the nonlinearity. (Notice that the large oscillations before the turbine trip, i.e. before 13 min, cannot affect the Suppression Pool temperature.)

The trend of peak pool temperatures is monotonic with respect to C_o , even though not all of the key parameters named in Section 4.1 above change monotonically. The steam discharge starts for all the above three cases at the time of turbine trip; it lasts longest (4.9 minutes) for the case with the largest C_o -value, less (4.3 minutes) for the case with the smallest C_o -value, and least (3.9 minutes) for the reference case. This nonmonotonic trend of valve closing time would call according to Eq.(1) for an ordering of peak pool temperatures which differs from that described above; the trend would have a minimum near the reference case. However, the fission power exceeds during the time of large oscillations the fission power of the non-oscillatory case (the case with the largest C_o -value). That leads according to Eq.(2) to higher pressures, the opening of additional valves and greater discharge of steam into the pool. Even though the fission power is only at first lower, and then, later near the valve

closing time, higher for the nonoscillatory case, and even though the valve closing time comes latest for the case with the largest C_o -value, the lower fission power early in the depressurization produces the lowest peak pool temperature for the case with the largest C_o -value. In general, the balance of effects from opposing phenomena determines the peak pool temperature.

(d) Initialization Effects. By TPG consensus, it had been decided to start all the transients for the response surface with the same initial pressure and power. The pressure is automatically controlled by the pressure-regulating system, and the power is adjusted by manual control-rod movement. These adjustments are necessary for simulations with variations of any of the parameters which affect the fission power, the fuel temperature, and the coolant flow.

The control-rod adjustment affects the amplitude and mean value of the fission power nonlinearly: the reduction in amplitude is very strong below, but less above, some threshold.

Figure 2.8 shows, for example, a fairly regular trend of peak pool temperature with respect to exit form loss coefficient, if the form loss coefficient varies between -5% and +30% of the reference value. Below -15%, the peak temperature drops sharply.

The simulation results show that the fission power amplitudes take 7.2 and 7.8 minutes, respectively, for the +15%, and the reference cases to reach twice the rated power, but only 9.8 minutes for the -15% case. The ensuing power oscillation amplitudes vary correspondingly. Consequently, the pressure history between the closing of the second and first valve banks, has a plateau which shortens gradually while the form loss coefficient decreases from +30% to 5%. At the values of -15% and lower, both banks close almost simultaneously, because the vessel energy is low. The nearly simultaneous closing of both SRV banks shortens the steam discharge duration suddenly; further lowering of the fission power reduces the peak pool temperature only insignificantly.

Varying the coefficient of core exit form loss produces two types of transients. The asymmetry is established before any vapor discharges into the pool. The same sharp change in slope of the peak-pool-temperature curve versus exit form loss coefficient may not have occurred if only the form loss had been changed, and the resulting change of initial power had been tolerated.

Figure 2.13 Asymmetric Responses of Reactor Pressure and Fission Power to Symmetric Variations of Drift Flux Parameter C_0 . (Middle Curve b is computed with reference C_0 , Top Curve a with $1.17 \cdot C_0$, Bottom Curve c with $0.83 \cdot C_0$).

(e) Explanation of Extrema

The trend of peak pool temperature versus SRV discharge flow rate, as shown in Figure 2.14, has extrema because the balance of effects from governing and mutually opposing phenomena is not monotonic.

Figure 2.14 shows twenty simulation results. Eight of the twenty runs were repeated; the repeat calculations confirmed all but one of the original calculations. All in all, thirty-one simulations were performed for the Suppression Pool temperature response to variations of SRV flow rate.

The dependence of the pool temperature response surface on the SRV flow rate is obtained by varying the valve cross-sectional area, so as to account for the uncertainty in predicting the critical flow through the valves. The variation of valve flow area does not affect any transient until the first bank of valves opens at the time of the turbine trip. Until that time, all simulations produce identical results. There is no manual control-rod movement needed to initiate the transients at the same power and pressure. In summary, the SRV flow rate dependence of the response surface is obtained with all other modeling parameters kept constant!

The peak pool temperature variation as a function of SRV flow rate depends predominantly and monotonically on the valve closing time t_c in Eq.(1). In fact, the peak pool temperature increases (with very little scatter) linearly with SRV closing time t_c , as shown in Figure 2.15.

However the valve closing time t_c , and therefore the peak pool temperature, are not changing monotonically with SRV flow rate, because an increase in SRV flow rate causes simultaneously

the reference case. The same is true for the various SRV flow decreases.

Moreover, the fission power change with SRV flow rate change is nearly compensated at times greater than two minutes after the turbine trip by the turbine bypass flow control system. At that time, the power is less than the Bypass capacity. The nearly compensated flow rate changes make the transients with SRV flow area variation threshold cases, where the valve closing time t_c responds by changing in both positive and negative directions to small SRV flow rate disturbances in one direction.

Finally, the continuous variation of SRV flow rate may, for example, increase the pressure continuously, while steam is being discharged through a bank of relief valves, until the pressure just reaches the next bank's opening setpoint, and the next bank just opens. The continuous pressure increase would require only an infinitesimally short opening time for the next valve bank, to produce a continuous depressurization in response to the continuous variation of SRV flow rate. Instead, the next valve bank remains open for an extended time, until the closing setpoint is reached. The associated depressurization is not continuous with respect to the SRV flow rate; crossing the opening setpoint of any SRV bank alters the transient discontinuously, because of the SRV's hysteresis.

The result is the nonmonotonic and possibly discontinuous variation of peak pool temperatures with variations of SRV valve flow rate, as shown in Fig. 2.14.

a.	more flashing,	⇒ more core voiding,	⇒ less power.
b.	coolant level reduction,	⇒ less core flow,	⇒ less power.
		⇒ less boron mixing,	⇒ more power.
		⇒ more RCIC injection,	⇒ more power.

Thus, the change of SRV flow in one direction alters the fission power in both directions of positive and negative change. The fission-power graphs obtained with various SRV increases show at different times in some cases more, and in other cases less power than the graph for

2.8 Statistical Analysis of Simulation Uncertainty

Technical Data Services (TDS) performed a statistical analysis of the set of peak suppression pool temperatures

Figure 2.14 Variation of Peak Pool Temperature as a Function of Safety and Relief Valve Flow Area.

Figure 2.15 Suppression Pool Peak Temperature vs. SRV Closing Time (t_e), Simulation of SRV Flow Rate Effects.

which were obtained from the 63 EPA simulations described above, and encompass the variations in 19 key parameters, as listed in Table 2.7. The purpose of the analysis is to estimate quantitatively the uncertainty in peak suppression-pool temperature, owing to the uncertainties in the above 19 parameters which were identified as being important in the PIRT of Section 2.4.3.

The statistical analysis is based on EPRI's Statistical Combination of Uncertainties (SCU) methodology [Doran et al. 1991] and was carried out with the aid of the PC program STARS.

The original objective was to produce a response surface spanned by low-order polynomials, as had been done earlier in the Large-Break LOCA CSAU project [Lellouche et al. 1990]. However, as explained in Section 2.7.4 above, the temperature of the suppression pool exhibits two characteristics that the software STARS [Doran et al. 1991] is unable to cope with, namely:

1. Nonsymmetric S-shaped data structures.
2. The presence of multiple extrema and one observed discontinuity with respect to the variation in the SRV flow rate (represented by the variation in SRV flow area).

As a result of these characteristics, quadratic regression for the response surface was found not to represent the data. The analysis was performed using only linear regression. The implications of this are discussed more fully in Appendix A.

The final conclusion of the analysis presented below is that the likelihood of exceeding the specified temperature limit of 352.59 K (175°F) in the suppression pool is probably zero for the postulated scenario.

However, the number of relevant simulation parameters is nineteen, while the software available at TDS is limited to only nine parameters. We inferred the above conclusion from the cumulative probability distribution of only the leading nine modeling parameters, and showed that the impact of the next ten modeling parameters would most likely not alter the above conclusion, even though the nineteen parameters were not statistically analyzed as a whole.

2.8.1 Parameter Analysis

In the STARS manual it is noted ([Doran et al. 1991], p.5-6) that the number of free coefficients, and therefore the number of necessary data points, for a P-th order polynomial surface of N arguments (parameters), is $(N+P)!/(N!P!)$. A quadratic surface ($P = 2$) of 19 parameters requires at least 210 data points to determine the 210 free coefficients. STARS is written for up to quartic ($P = 4$) polynomials, but limited to a maximum of 9 parameters and it contains only a partial cubic and quartic capability. This makes it impossible to deal simultaneously with all 19 parameters.

The statistical analysis was therefore carried out as the 17 Cases listed in Table 2.8 below, by using the order of the parameters given in Table 2.7. The results of these Case studies may be found in Appendix A.

For Table 2.8, a parameter is considered "worse" than another if its highest calculated suppression pool temperature is greater than that of the other. In this sense the 19 modeling parameters may be listed in the order of decreasing maximum suppression pool temperature, as shown in Table 2.9 below.

Table 2.8 Case List

Case No.

1. First nine parameters
2. Second five parameters
3. Last nine parameters
4. Last four parameters
5. Worst single parameter
6. SRV parameters
7. Case 6 + feed water controller gain
8. Case 7 + vapor generation
9. Case 8 + drift flux
10. Case 9 + condensation in downcomer
11. Case 10 + heat transfer coefficient
12. Case 11 + fuel gap conductance
13. Case 12 + boron worth
14. Last nine parameters less the SRV flow area parameter
15. Seven worst parameters
16. Eight worst parameters
17. Nine worst parameters

It should be pointed out that the maximum value of the carry-under fraction occurs at the nominal value of zero. The most interesting aspect of Table 2.9 is that except for the SRV Pressure Set Point effect, no other parameter causes as much as a 1.7 K (3°F) change in the suppression pool peak temperature from the nominal value of 329.27 K (133.02°F).

Indeed, the square root of the sum of the squares (SRSS) of the maximum deviations from the nominal value, gives an absolute maximum of 350.69 K (171.57°F) which is less than the nominal licensing peak temperature of 352.59 K (175°F).

Of the 17 cases studied as linear surfaces, only Cases 15 through 17, combining the sets of 7, 8, and 9 worst

parameters, show any temperatures above 352.59 K (175°F), with probabilities of exceeding 352.59 K (175°F) of $2 \cdot 10^{-4}$, $1.3 \cdot 10^{-4}$ and $5 \cdot 10^{-4}$, respectively.

However, for the quadratically fit surfaces, the maximum bin is 357.59 K (184°F) and the maximum probability of exceeding 352.59 K (175°F) is 0.00988 for Case 12 (note that Case 12 as a linear fit shows a maximum bin of 350.93 K (172°F)).

Because of the difference between the linear and quadratic response surfaces for Case 12, the effect of omitting the quadratic terms of the SRV parameters from the response surface representation was evaluated.

**Table 2.9 Modelling Parameter Ranking
According to Highest Suppression-Pool Temperature Produced**

Parameter No.	Parameter	Highest Temperature Calculated, °F
15	SRV Pressure Set Point	170.8
2	Inlet Loss Coefficient	135.96
4	Doppler Coefficient	135.93
1	Outlet Loss Coefficient	135.71
16	SRV Flow Area	135.61
10	Heat Transfer Coefficient	135.2
7	Feedwater Heating	135.14
14	Feedwater Controller Gain	135.13
6	Axial Peaking Factor	134.88
18	Level Control Set Point	134.58
12	Drift Flux	134.50
3	Void Coefficient	134.35
13	Vapor Generation	134.32
5	Radial Peaking Factor	134.30
17	Level Control Time Constant	133.8
8	Boron Worth	133.54
9	Fuel Gap Conductance	133.52
11	Condensation in the Down Comer	133.3
19	Carry Under Fraction	133.02

Table 2.10 shows the importance of the quadratic terms for pressure set point and flow rate uncertainties.

The first thing to note is that the nonsymmetrical nature of the probability distribution function, as indicated by the difference between the median and the mean, virtually disappears when the SRV Pressure Set Point response is modeled linearly. The probability of exceeding 352.59 K (175°F) almost vanishes as the quadratic terms are removed fully from both the SRV parameters.

This indicates that the probability of exceeding 352.59 K (175°F) results largely from fitting the SRV parameter response with a quadratic polynomial. It is shown in Appendix A that the quadratic fit introduces extrema that cannot be justified by the data in Table 2.7. It is also shown in Appendix A that certain artifacts of the regression call into question the use of quadratic elements for any of the parameters, and that the linear regression appears to be robust to small variations in input data,

while the quadratic regression surface appears to amplify small variations in input data.

2.8.2 Probability Distribution Functions

Figures 2.16 through 2.19 show the probability distribution and cumulative probability distribution function, as obtained with 50,000 and 100,000 case histories, both for Case 17, the case with the 9 "worst" parameters.

It is clear that there are still some problems near the mean, as one would have thought that the pdf would have been smoother at 100k histories. Such irregularities are, however, unlikely to alter the gross statistics at all. This may be seen by noting that the cumulative distribution function is nearly unchanged between the 50,000- and 100,000-history computations, and that the important

Table 2.10 Effect of the Quadratic Term in the SRV Parameter Fitting for Case 12 (50,000 History Studies)

	RMS	Mean	Median	Standard Deviation	Maxi. Bin	Prob (T > 352.6K)
Case 12 (Base) SRV Parameters by quadr. repr.	6.16	137.2	133.7	16.4	184	0.00988
SRV Press. Set Point linear	8.42	134.2	134.3	18.0	182	0.0029
SRV Press. Set Pt. & Flow Rate linear	9.21	135.9	135.9	16.5	178	0.00033

Table 2.11 Cumulative Function Percentiles for Case 17

Percentile Temperatures	Number of Histories	
	50k	100k
T ₂₅	121.2	121.1
T ₅₀	134.2	134.1
T ₇₅	147.2	147.3
T ₉₅	160.2	160.3
T ₉₉	166.9	166.9

percentiles are not significantly changed, as seen in Table 2.11 above.

Finally in Table 2.12 below, we show the cumulative function percentiles computed with 50,000 histories for Cases 15, 16, and 17, that is for the combinations of seven, eight, and nine "worst" parameters.

It is clear, then, that 352.59 K (175°F) is at the far tail of the cumulative probability distribution, and the error in estimating the associated probability is rather large. It is, however, certain that the probability is less than one percent since the T₉₉ value is less than 349.82 K (170°F).

Table 2.12 Effect of Increasing Number of Parameters on Cumulative Percentiles

Percentile Temperatures	Number of Parameters ¹		
	7 "Worst"	8 "Worst"	9 "Worst"
T ₂₅	121.6	121.3	121.2
T ₅₀	134.6	134.2	134.2
T ₇₅	147.6	147.3	147.2
T ₉₅	160.1	160.0	160.2
T ₉₉	166.3	166.2	166.9

¹ "Worst" means: producing the highest maximum suppression pool temperature.

Figure 2.16 Probability Distribution Function for Suppression Pool Peak Temperature, 50,000 Case Histories.

Figure 2.17 Cumulative Probability Distribution Function for Suppression Pool Peak Temperature, 50,000 Case Histories.

Figure 2.18 Probability Distribution Function for Suppression Pool Peak Temperature, 100,000 Case Histories.

Figure 2.19 Cumulative Probability Distribution Function for Suppression Pool Peak Temperature, 100,000 Case Histories.

2.8.3 Conclusion from Statistical Analysis

We conclude from our calculations that:

1. It is inappropriate to model the data in Table 2.7 using a surface containing a quadratic term.
2. In all likelihood, the probability of exceeding 352.59 K (175°F) is zero for the postulated scenario. However, because of our inability to model all 19 parameters simultaneously, Case 17 for the 9 most important parameters may be presumed to bound the result with the following statement

$$\Pr(T > 353 \text{ K}) < 5 \times 10^{-4}$$

At 100,000 histories this would be reduced to $< 3 \times 10^{-4}$. It must be noted that the square root of the sum of the squares of the maxima of all 19 variables is only 350.69 K (171.57°F) which would imply the correctness of our view that $\Pr(T > 353 \text{ K}) = 0$ (even if we cannot validate it further).

3. Conclusions

1. The final conclusion of the uncertainty analysis performed here along the guidelines of the CSAU methodology [Boyack et al. 1990] is that the likelihood of exceeding the specified temperature limit of 352.59 K (175°F) in the suppression pool is probably zero for the postulated BWR ATWS scenario.

The NRC-postulated ATWS scenario is initiated by a trip of both Recirculation Pumps, and then develops limit-cycle power and flow oscillations. The oscillations are disrupted by a turbine trip, which is accompanied by the postulated failure of the MSIVs to close at low level. Operator actions to reduce the reactor pressure (activation of low-low SRV pressure set points) and the mixture level in the downcomer (control of feedwater injection), and boron injection shut off the fission power.

2. Bases. The above conclusion is reached on the basis that the extreme estimate of total uncertainty, which means the square root of the sum of squares of the maximum pool temperatures, computed with the uncertainties of all nineteen modeling parameters, is only 350.7 K (171.6°F).

The probability of the Suppression Pool temperature to exceed the specified limit of 175°F is estimated to be

$$\Pr(T > 353 \text{ K}) < 5 \times 10^{-4}.$$

This estimate is based on the computed cumulative probability distribution that was obtained with 50,000 histories from the nine most important (out of nineteen) modeling parameter uncertainties.

3. Nonlinearities. The reactor system response to parameter variations is extremely nonlinear under the reference conditions, because the system is unstable (oscillating), yet close enough to the stability border that the change in one direction of one in a class of modeling parameters renders the reactor stable (non-oscillatory), while the same change in the opposite direction of the same parameter renders the reactor only slightly more unstable (with slightly greater oscillation amplitudes).
4. Extrema. The trend of peak pool temperature versus SRV discharge flow rate, as shown in Figure 2.14, has extrema because the balance of effects from governing and mutually opposing phenomena is not monotonic.
5. Discontinuities in the trend of peak pool temperature versus SRV discharge flow rate, (shown in Figure 2.14) are caused by hysteresis in SRV valve actions (different pressure setpoints for valve opening and closing).
6. Because of Findings 4 and 5 above, it is inappropriate to fit the data (in Table 2.7) for the response surface with quadratic terms. Linear regression was used instead for the statistical analysis.
7. The uncertainty of predicting the suppression-pool peak temperature, measured as the difference between the mean and 95-percentile temperatures is 14.44 K (26°F). This is seen in Table 2.10. The uncertainty is dominated by the uncertainty bracket of $\pm 10\%$ in measuring the SRV Mass Flow Rate.

4. References

- Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wilson, G.E., Wulff, W. and Zuber, N., "An Overview of the Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology," *Nuclear Engineering and Design*, Vol. 119 No.1, 1990.
- Doran, K. J., May, R. S. and Engel, R. E., "STARS: Statistical Analysis by Response Surface, Theory and User's Manual," prepared by S. Levy Incorporated, for Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto CA 94304, NP7558-CCML, January 1992.
- Ishii, M., "One-Dimensional Drift-flux Model and Constitutive Equations for Relative Motion Between Phases in Various two-Phase Flow Regimes," ANL-77-47, Argonne National Laboratory, Argonne, IL, 1977.
- Lellouche, G.S., Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Levy, S., Rohatgi, U.S., Wilson, G.E., Wulff, W. and Zuber, N., "Uncertainty Evaluation of LBLOCA Analyses Based on TRAC-PF1/MOD1," *Nuclear Engineering and Design*, Vol. 119 (1990) No.1, 1990.
- Technical Program Group: Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wilson, G.E., Wulff, W. and Zuber, N. Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability and Uncertainty Evaluation Methodology to a Large-Break Loss of Coolant Accident, NUREG/CR-5249, Idaho National Laboratory, Idaho Falls, ID 83415, 1989.
- Watkins, J. C., Ghan, L. S. "AHP Version 5.1 Users' Manual," EGG-ERTP-10585, EG&G Idaho, Inc., 1992.
- Wilson, G.E., Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wulff, W. and Zuber, N., "Characterization of Important Contributors to Uncertainty," *Nuclear Engineering and Design*, Vol. 119, No.1, 1990.
- Wulff, W., Cheng, H.S., Lekach, S.V., and Mallen, A.N. "The BWR Plant Analyzer," Final Report, NUREG/CR-3943, Department of Nuclear Energy, Brookhaven National Laboratory, Upton NY 11973, 1984.
- Wulff, W., "Uncertainty in Modeling and Scaling in the Prediction of Fuel Stored Energy and Thermal Response," NUREG/CR-5232, Department of Nuclear Energy, Brookhaven National Laboratory, Upton NY 11973, 1988.
- Wulff, W., Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wilson, G.E. and Zuber, N., "Assessment and Ranging of Important Parameters," *Nuclear Engineering and Design*, Vol. 119, No.1, 1990.
- Wulff, W., Cheng, H.S., Mallen, A.N. and Rohatgi, U.S. "BWR Stability Analyses With The BNL Engineering Plant Analyzer," NUREG/CR-5816, Brookhaven National Laboratory, Upton, NY 11973, 1991.
- Zuber, N., Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wilson, G. E. and Wulff, W., "Evaluation of Scale-Up Capabilities of Best Estimate Codes," *Nuclear Engineering and Design*, Vol. 119, No.1, 1990.

Appendix A: Details of Statistical Analysis (G. S. Lellouche)

A.1 Statistical Analysis Input Data Problems

As an example of the difficulty of dealing with sigmoidal data structures we consider the most important parameter: the SRV Setpoint variation (see top entry in Table 2.9). The following table shows four separate fittings and Monte Carlo analyses of this parameter as we vary it by $\pm 5\%$ of the nominal value. The maximum calculated temperature is 170.8°F.

The Monte Carlo analysis was performed with 50,000 histories. It is interesting, that the introduction of a quadratic term, to either a linear or cubic root function, introduces extrema which, in turn produce Monte Carlo hits greater than any of the input data [Lellouche et al.]. Accepting a fit that exceeds the maximum of the input data, without other information justifying the existence of such extrema inside the range of parameter variation is improper. We note that the two extrema which appear with the full cubic are at 0.0309 and -0.0434 on the Setpoint parameter range of (-0.05, 0.05), and that data were taken at -0.05, -0.02, 0.0 and 0.05. It would appear that the fittings which contain quadratic elements should be excludable. We note also that the odd surfaces produce distribution functions with mean and median values close together while those with quadratic terms produce skewed distributions.

In order to interpret the data set containing multiple extrema we consider a combined analysis of the SRV Setpoint and Flow Area data. The data from the variation of flow area appear to be the result of the pressure-dependent dead-band in the opening and closing actuator of each SRV valve bank. This dead-band introduces a nonlinearity that is the origin of the non-monotonic behavior seen in Fig. 2.14. We must treat these data therefore with a bounding analysis.

The SRV Flow Area results are given in Table 2.7 as data points 39-56; the calculations cover the parameter range of (-0.1, 0.1) of the full throat opening. These data show at least seven extrema and one of them is essentially a discontinuity. We replace the 18 data by the values at (0.1, 0.03, -0.03, -0.045, -0.1). These values produce a bounding curve with a higher average than the complete set. This bounding curve is used in the analysis.

Then we consider also a second bounding set at (0.1, -0.1) and a third set at the same locations but with both temperature values at 134.78 (the higher of the two values). We determine with these bounding SRV pressure setpoint data and the SRV flow rate data, the effect that the flow area variation has on peak pool

Table A-1 Analysis of the SRV Setpoint Results
(50,000-History Study)

Regression Fitting	RMS	Mean	Median	Standard Deviation	Maximum Bin
Linear	8.96	144.0	144.0	13.9	169
Quadratic	9.59	142.1	139.1	14.6	173
Cubic (w/o square)	11.78	142.6	142.6	17.5	167
Full Cubic	0.00	137.8	132.9	23.5	173

Appendix A: Details of Statistical Analysis

**Table A-2 Effects of the SRV Flow Area Variation
(50,000 History Study)**

Case No.	RMS	Mean	Median	Standard Deviation	Maximum Bin
1	9.02	137.2	137.2	14.7	166
2	6.16	138.1	134.0	15.2	174
3	11.4	139.5	139.6	14.6	170
4	9.59	139.2	136.4	14.9	174
5	9.52	140.7	140.7	14.3	166
6	9.59	140.8	137.6	14.6	174

temperature. The statistics are found in Table A-2. Six Cases were studied, encompassing the following types of fits:

- 1: Setpoint and Flow Area data as a 5 point bound using a linear fit.
- 2: Case 1, using a quadratic fit.
- 3: Case 1, (0.1, -0.1) data only for the Flow Area.
- 4: Case 3 as a quadratic.
- 5: Case 3, using a pseudo data point at 0.1 of 134.78°F (the same as at -0.1).
- 6: Case 5 as a quadratic.

As the peak pool temperature average resulting from the flow area variation increases, (Cases 1, 3, 5 and 2, 4, 6) the mean and median increase also, as would be expected. However, the Maximum Bin does not change appreciably; it increases slightly and then decreases for the linear fit and remains the same for the quadratic. Using the five point bound for the Flow Area data appears reasonable since using even higher bounds does not produce a significant effect.

A.2 Evaluation of Quadratic and Linear Regression Fits

Table A-3 below shows the summary of all the studies which are the basis of the statistical analysis in this report.

The most important question arising from Table A.3 is "Why has the $P(T > 175)$ jumped nearly two orders of magnitude between Cases 11 and 12?" Case 12 is Case 11 with the addition of the Fuel Gap Conductance parameter, which is 17th on the list of important parameters showing an increase of only 0.5°F over the nominal case at its high end.

To understand why this has occurred, we examine the regressions for Cases 11 and 12. In particular, we note that the prediction of the nominal point for Cases 11 and 12 are both high by $\approx 2.5^\circ\text{F}$. This means that the curve joining the peak pool temperature predictions as functions of the Fuel Gap Conductance (FGC) predictions, with the estimated nominal point has negative curvature. This increases the values of the mean and median. If the curvature were positive, the mean and median would move to lower values. This can be observed in Table A-4 where we vary the values of the FGC end points from 132.02°F to 136.02°F. Since the nominal point on the response surface is 135.59°F, the curvature moves from negative to positive, and the results match Case 11 as the varied values pass the nominal point of 135.59°F.

The same random seed of (98777) is used for all entries in Table A-4, to avoid purely statistical distortions. The reason why the "as Given" case differs from that in Table A-3 is that a different starting seed was used.

It is noted that the tail of the distribution presented as a probability varies by two orders of magnitude over the

Table A-3 Summary of Statistical Results

Case No.	Fit	No. of Histories(k)	RMS	Mean	Median	Standard Deviation	Max. Bin	Prob (T > 175) x 10 ⁴
1	Quad	50	1.54	132.2	132.5	6.22	151	0
		100	1.54	132.2	132.5	6.19	151	0
2	Quad	50	0	134.5	134.8	3.37	142	0
	Quad	100	0	134.5	134.8	3.38	142	0
	Linear	50	1.32	134.4	134.4	3.14	144	0
3	Quad	50	5.57	126.2	122.4	16.05	171	0
	Linear	50	6.85	129.2	129.2	15.51	168	0
4	Quad	50	1.08	126.7	127.4	4.20	136	0
	Linear	50	2.32	130.3	130.3	2.24	137	0
5	Linear	50	8.96	144.0	144.0	13.9	169	0
	Linear	100	8.96	144.0	144.0	13.9	169	0
	Quad	50	9.59	142.1	139.1	14.6	173	0
	Quad	100	9.59	142.0	139.0	14.6	173	0
	Cubic	50	11.78	142.6	142.6	17.5	167	0
	Full Cubic	50	0	137.8	132.9	23.5	173	0
6	Linear	50	9.02	137.2	137.2	14.7	166	0
	Quad	100	6.16	138.1	134.0	15.2	174	0
	Quad	100	6.16	138.1	134.0	15.2	175	0
7	Linear	50	8.99	136.2	136.2	15.0	168	0
	Quad	50	6.16	136.7	132.7	15.4	176	2.3
	Quad	100	6.16	136.6	132.7	15.4	176	1.2
8	Quad	50	6.16	134.7	130.9	15.7	176	2
9	Linear	50	8.48	134.8	134.9	15.4	172	0
	Quad	50	6.16	133.6	129.7	15.8	176	2.2
10	Linear	50	8.07	134.5	134.5	15.5	172	0
	Quad	50	6.16	132.7	129.0	15.9	176	1.3
11	Linear	50	7.81	134.9	134.9	15.6	172	0
	Quad	50	6.16	132.7	129.1	16.1	176	3
12	Linear	50	7.5	134.6	134.6	15.6	172	0
	Quad	50	6.16	137.2	133.7	16.4	184	99
13	Linear	50	7.5	134.4	134.4	15.5	174	0
	Quad	50	6.16	136.2	132.8	16.4	184	67

Appendix A: Details of Statistical Analysis

Table A-3 Summary of Statistical Results (Cont'd.)

Case No.	Fit	No. of Histories(k)	RMS	Mean	Median	Standard Deviation	Max. Bin	Prob (T > 175) x 10 ⁴
14	Linear	50	7.67	129.1	129.1	15.5	165	0
	Quad	50	7.42	123.6	120.6	15.8	168	0
15	Linear	50	6.88	134.7	134.6	16.1	178	2
	Quad	50	4.73	134.2	136.5	16.6	182	45
16	Linear	50	6.29	134.2	134.2	16.2	178	1.3
	Quad	50	4.31	133.8	129.9	16.8	184	54.
	Quad	100	4.31	133.7	130.0	16.7	184	53.
17	Linear	50	6.18	134.2	134.2	16.4	182	5
	Linear	100	6.18	134.2	134.1	16.4	180	3

Table A-4 Interpretation of the Differences Between Cases 11 and 12 (50,000 History Study)

FGC Values	Mean	Median	Pr(T > 175°)	T ₉₅	Predicted Nominal
As Given	137.3	133.8	0.0103	167.4	135.59
132.02	137.1	133.7	0.0102	167.3	135.59
133.02	135.9	132.4	0.0054	165.8	135.59
134.02	134.7	131.1	0.0023	164.4	135.59
135.02	133.4	129.8	0.0007	163.2	135.59
136.02	132.2	128.6	0.0001	160.0	135.59

**Table A-5 Consideration of the Differences Between Cases 11 and 12
Based on Linear Regression**

FGC Values	Mean	Median	Pr(T > 175°F)	T ₉₅	Predicted Nominal
As Given	134.7	134.8	0	159.0	134.12
132.02	134.0	134.1	0	158.3	134.11
133.02	134.0	134.5	0	158.8	134.16
134.02	134.9	134.9	0	159.2	134.22
135.02	135.3	135.4	0	159.6	134.28
136.02	135.8	135.8	0	160.0	134.34

range of the study, while the 95th percentile temperature changes by only 7°F. This merely points out the known difficulty of statistically determining accurately the probability of exceeding a specific value as opposed to determining the temperature at a specific (non-infinitesimal) probability.

The results in Table A-4 are of concern since they depend on the curvature of the actual data with respect to the predicted nominal point, however for all the cases in Table A-2 the peak pool temperatures of the FGC data other than the nominal point are always predicted with an error of less than 0.05°F, and generally less than 0.01°F).

In order to further nail down the problem, we repeat this study with a linear regression. These results (based on a random seed of 58934) are found in Table A-5.

Here we find a satisfying and robust set of results. Although the FGC values are varied by 4 degrees, the mean, median, and T₉₅ vary by less than 2 degrees. For the quadratic regression the same 4 degree variation produced a more than 5 degree variation in the mean, median, and T₉₅ values. That is, the quadratic regression appears to be unstable with respect to variations in the input data (yielding results greater than the variation) while the linear regression yields results of only half the variation. We conclude from the comparison of the results in Table A-4 and A-5 that the quadratic regression is not stable to small variations in the input data, but that the linear regression is robust to such variations.

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