TOPICAL REPORT SUMMARY

Overview of Topical Report Development

The Electric Power Research Institute (EPRI), in conjunction with Exelon and other EPRI member utilities, plan to develop a Topical Report for implementation of Data Validation and Reconciliation methodology[1], herein referred to as DVR, as a means to reduce the uncertainty of determinations of nuclear plant Core Thermal Power.

The approach to the Topical Report development shall be to establish the technical basis for DVR and perform related evaluations to substantiate the uncertainty claims of the DVR process. Evaluating the DVR process for performing Measurement Uncertainty Recapture (MUR) power uprates will include performing failure modes and effects analyses (FMEA) to identify all areas of possible errors in the results and objective justification for self-identification of process failure.

Portions of this report have been derived from the EPRI report "Process Data Reconciliation User Guideline", EPRI 3002013197.

The following Utilities, Institutions and Power Industry service providers have joined with EPRI in the development of the Topical Report:

Exelon - Fleet
Duke Energy – Fleet
Ameren Callaway Station
Luminant- Comanche Peak Station
FENOC – Fleet
Palo Verde Nuclear Generating Station
Xcel Energy – Monticello, Prairie Island
South Texas Project
Belsim International
Texas A&M Nuclear Engineering Department

The remainder of this paper provides a suggested outline and brief description of the various project tasks associated with the development of the Topical Report.

[1] Methodology and technology are used interchangeably throughout this topical summary.

Topical Report Outline

1.0 Introduction

Core Thermal Power (CTP) determination in the nuclear power industry has typically relied on the accurate measurement of various inputs to the power calculation, in particular feedwater flow. These input measurements are often single element measurements and therefore a failure of the instrumentation will result in an error of the power calculation. As described in the Operating Experience (OE) section of this document there have been numerous failures experienced in the nuclear industry due to this reliance on only a few instruments to determine core thermal power. The Data Validation and Reconciliation (DVR) process provides an opportunity to include significantly more instruments to the evaluation of core thermal power thus resulting in a more robust approach. Use of a statistical approach to monitor a nuclear power plant's safety functions have been commonly used in the PRA (Probable Risk Assessment) process and is also a factor with the new 10 CFR 50.69 process.

Measurement Uncertainty Recapture (MUR) power uprates have been performed domestically with the use of ultrasonic flow meters. However, even with the use of these more accurate flow meters the vulnerability of a single point measurement still exists.

Data reconciliation methods have been used in Europe to provide corrections to core thermal power measurement. The DVR approach uses analytical thermodynamic principles and measurement uncertainty analyses that allows for the incorporation of additional plant instrumentation.

Use of DVR technology offers the nuclear power industry plant a method of improving the reliability of CTP calculations by reducing single point measurement vulnerabilities. DVR may also provide a more accurate measure of CTP as the method is based upon a statistical reconciliation of the plant instrumentation with the plant's actual operating condition.

DVR can be used to detect potential metering problems with existing plant feedwater flow metering (UFMs, venturis, nozzles, etc). DVR may be also used to supplement plant feedwater flow metering (to improve the statistical accuracy of the flow measurement by combining the values from DVR and the plant metering. DVR can be useful for detecting problems with other instrumentation related to the CTP, or any other instruments that are included in the DVR model.

Development of the Topical Report for MUR will use experience gleaned from Appendix K uprates.

1.1 General Description

The German VDI-2048 engineering standard was developed as an industry code for balance of plant turbine acceptance testing, much like the ASME PTC-6 code used in the United States. The objective of the VDI code is to reduce the acceptance testing complexity and cost by using a higher quantity of lower cost test instrumentation in diverse locations. The code introduces the concept of using an empirical covariance matrix and Gaussian corrections as a means to correct errors with the test instruments and assess the quality of the test measurements and results. The methods provided in the VDI code can be applied to the plant CTP calculation using existing plant instrumentation.

ASME 19.1-2013 Section A.2 describes a general methodology for determining a weighted overall uncertainty whenever a value of a parameter is approximated by several different measuring methods. However, the ASME codes do not address use of empirical covariance matrices and Gaussian corrections as a means determining instrumentation and test uncertainties. Therefore, at this time, VDI-2048 is the primary source for the engineering and statistical calculations as related to the DVR process.

As part of this project, applicability of the DVR method to the ASME codes will be investigated.

1.2 Comparison with Other Different Methodologies

The CTP at most nuclear plants is calculated via a heat balance calculation that utilizes measurements of steam generator (PWRs) or reactor feedwater flowrates (BWRs). The feedwater flowrate is the main error contributor to the CTP calculation. Many plants use feedwater flow metering that consists of venturis, nozzles, or UFMs. Each of these *single point* methods may be prone to specific error conditions.

Use of DVR technology provides a method of improving the reliability of CTP calculations by reducing the dependence on single point measurement vulnerabilities. DVR can also reduce the uncertainty of CTP calculations as the method is based upon a statistical reconciliation of other CTP related plant instrumentation with the plant's actual operating condition. In addition, the DVR CTP calculation can be more robust as it uses a number of plant measurements (typically 80 to >200) and is less vulnerable to single point instrument errors.

1.3 Comparison with Other Similar Methodologies

DVR utilizes numerical methods software to calculate a value of plant reactor CTP and other plant parameters using a number of plant instruments. The DVR technology is unique in that it couples thermodynamic first-principles based engineering and physics calculations, system, cycle mass and energy calculations with statistical methods. Alternatively, there are a variety of numerical methods and software that have been used by electric power utilities and other industries to identify plant instrumentation measurement errors. Potentially, some of these methods could be used to calculate the CTP as an alternative method to DVR. Alternate numerical methods technologies will be investigated and compared to DVR. Some of the numerical methods are given as follows.

1.3.1 Independent Component Analysis (ICA)

ICA has been used for facial and pattern recognition. ICA is a computational method for separating a multivariate signal into additive subcomponents.

1.3.2 Multivariate State Estimation Technique (MSET)

MSET has been used by the Department of Energy as a signal validation method at the Florida Power Corporation Crystal River 3 nuclear plant. MSET uses State estimation models, “trained” using sampling of plant data and a Fault detection model that performs statistical tests.

1.3.3 Non-Linear Partial Least Squares (NLPLS)

NLPLS has been developed as an improvement over traditional regression analysis where strong non-linear relationships may exist between different data sets in the process under review. NLPLS has been primarily used in the chemometrics and chemical process industries.

1.3.4 Auto-Associative Neural Networks (AANN)

AANNs have been used for image recognition in computer and robotic vision systems. Use of neural network technology for detection of errors with nuclear plant feedwater flow instrumentation, and the CTP, have been investigated by the University of Tennessee Department of Nuclear Engineering. NNs are “trained” using samples of plant data.

1.4 Industry Experience with DVR Methodology

DVR software has been used by the nuclear power industry in the U.S. and Europe to assess turbine cycle thermal performance, balance of plant feedwater flow metering and accuracy of the plant calorimetric, or measure of the plant’s licensed reactor power.

DVR technology has been used by Kernkraftwerk at the Leibstadt plant for power recovery from feed nozzle fouling. The DVR technology used at Leibstadt allows compliance with VDI-2048.

At the Kernkraftwerk Gundremmingen plant in Germany, the DVR technology has been used to correct the venturi feedwater flow metering for measurement errors. The method has been approved by the local regulators. The MUR process has also been initiated at NPP Gundremmingen B and C with approval of TÜV Süd (independent certifying body) and the Bavarian Environmental Ministry (Regulator).

At the Beznau nuclear power plant in Switzerland, DVR technologies have been used since 2004 and have replaced use of radioactive sodium tracer testing as a means of feedwater flow metering calibration.

The Eletronuclear - Angra nuclear power station uses DVR to correct the plant CTP for plant instrumentation errors.

DVR technologies have also been used to assess the balance of plant thermal performance, perform turbine acceptance testing, and detect errors with plant instrumentation at the Gösgen and Leibstadt plants in Switzerland, the Forsmark plant in Sweden, and the Philippsburg plants in Germany.

The Borssele nuclear plant in the Netherlands also uses DVR technology and is currently in the process of licensing the plant for MUR use.

In the United States, DVR technology has been used at several sites with Operating Event issues due to suspected errors with the ultrasonic feedwater flow metering within the plant licensed calorimetric (CTP). This section captures current applications of the technology in the power industry in general and specifically in the Nuclear Power industry. These implementations of DVR technology will be discussed in the Topical Report.

1.4.1 Beneficial Features of DVR Methodology

- Utilizes many plant instrument inputs, so less sensitive to single point failures
- Provides detailed statistical analyses of the plant measurements.
- Details of the DVR calculations can be reviewed so that it is apparent how DVR arrived at the results.
- Provides calculations based on sound, commonly accepted, engineering and statistical principles.

1.4.2 Reason for Topical Report

In the two decades since the MUR uprates have been implemented, there have been numerous down powers by the plants due to an instrument failure, usually on the part of the ultrasonic flow meter or venturi/nozzle fouling. There have also been instances where plants have operated higher than their core thermal power license limit due to issues with the plant feedwater flow metering (see the section on OE below). In many cases, these conditions are due to a failure of a single instrument.

1.4.2.1 INPO Report

The increasing number of incidents involving ultrasonic flow meters prompted INPO to commission a Topical Report TR4-34, “Review of Feedwater System Ultrasonic Flowmeter Problems”, which was published in March 2004. The report concluded that:

- There is an increased trend over the past four years in the number of events involving ultrasonic flow meters used to calculate reactor power. This trend is a concern because many of the events resulted in reduced margins to safety. This indicates that users of this technology need to carefully monitor instrument output and closely oversee the installation of new systems.

- Reactor power indication was directly affected in 10 of the 14 reported events over this period. Reactor power limits (100 percent) were exceeded during seven events with one station operating slightly in excess of its 102 percent analyzed limit for approximately 15 months. For example, a hypothetical 1000 MWe nuclear plant operating on a 1.7% MUR would have to derate 17 MWe to go back to the original 2% uncertainty. In a 24-hour period, this would be 408 MW-hrs, costing the plant about $18,400 at $45 MW-hr. If the de-rate lasted several days, this would amount to a serious revenue loss to the utility.

- Both flow meter types experienced problems during the analyzed period. Four events involved the use of the cross-correlation flow meter during which reactor power limits were exceeded. One of these stations operated above 100 percent reactor power for more than three years. Transit time flow meters (also called Leading Edge Flow Meters) were involved in the remaining ten events. Three of these events reported operation above 100 percent reactor power.

- Nine of the 14 reported events involved human error. Seven of the nine involved some type of error, oversight or lack of knowledge on the part of the vendor.

- There may be an over-reliance on vendor expertise. More detailed review and questioning by station personnel pertaining to the basis for software programming and post modification testing, as well as acceptance of post installation test results, are required.

1.4.2.2 Operating Experience

CLINTON STATION

Venturi fouling (and de-fouling) became particularly evident after C1R07 (November 2000) when zinc injection was initiated. An Extended Power Uprate (EPU) was implemented following C1R08 (March 2002). Post-EPU operation data showed that the indicated power was approximately 1% higher than expected at the fixed generator output and operating conditions (i.e. back-pressure). This was confirmed in 2002 via temporary ultrasonic instruments and was ultimately determined to be due to feed water venturi fouling.

A 10 CFR 50.59 analysis in 2002 allowed Clinton to implement a 1% correction factor to correct the indicated venturi measurements. The feedwater venturi started de-fouling (cleaning) itself after hydrogen injection was initiated in July 2002. By the summer of 2003 Clinton recognized the flow values were moving in a non-conservative direction and the 1% correction factor was removed in early 2004. The extent of venturi fouling has varied since 2004 and increased significantly during the summer of 2011.

BYRON AND BRAIDWOOD STATIONS

In 1999, Exelon implemented installation of Crossflow ultrasonic flow meters (UFM) at Dresden, Byron, Braidwood and LaSalle plants. The first plant the Crossflow UFM was installed at was Dresden.

Byron and Braidwood are for all purpose's identical plants. There is no significant difference between them. PEPSE thermal performance computer models were created for these plants. The PEPSE models at both plants were predicting no shortfall of power due to overly conservative feedwater flow measurement.

An investigation spanning over a 5-year period finally found that there was an ultrasonic wave generated by the feedwater regulating valves that influenced the Crossflow meter readings. Both plants had operated for a significant period using a non-conservative flow rate to calculate reactor power. The end result was that the Crossflow meters were removed from all three plants and they all operated on uncorrected nozzle/venturi flows until the LEFM meters were installed. Due to the problems discovered at Byron and Braidwood, Crossflow UFM was never installed at LaSalle.

ST. LUCIE STATION

St. Lucie Unit 2 average power level was calculated to have been 100.1% power for greater than 8 hours on September 25, 2000. This power level resulted from an instrumentation failure on the LEFM which was utilized by the DDPS (Digital Data Processing System) Plant Computer Calorimetric power level indication. A failure in the DP1 transducer path on the 'A' Loop of the LEFM instrument resulted in a decreasing feedwater (FW) flow output value. This resulted in a non-conservative feedwater flow rate being used in the reactor power calculation and the plant operated for more than 8 hours at 100.1% licensed power. Operations reduced power immediately upon discovery. The LEFM was repaired and put back in service.

H.B. ROBINSON STATION

On July 13, 2003, the plant process computer indicated a "Power Limit Warning" alarm. The LEFM power indication showed an increase from 2337 MWth to a peak of 2346.73 MWth. Redundant power indications did not show an actual power increase, however, Operations conservatively adjusted power down approximately 5 MW net while an LEFM investigation was performed.

Investigations into the incident indicated that the Acoustic Processing Unit (APU) had failed. The card was replaced and the faulty unit was sent to the vendor for study. Analysis indicated that the card was functioning properly, but there was an error on one line of the code that operated it. A software patch was implemented to correct the error, which was determined to be the cause of the erroneous flow reading.

BEAVER VALLEY STATION

On July 10, 2001, the commissioning test for the newly installed LEFM revealed that the Beaver Valley Unit 2 had been running at approximately 101.5% power using the venturis.

Another incident on August 8, 2002 happened when the LEFM indicated again that reactor power was too high. An investigation by Caldon revealed that a recent software upgrade had errors in it that caused the failure.

Another LEFM failure was experienced on the Unit 2 LEFM on July 9, 2013.

The root cause was determined to be a defective circuit (LEFM 'A' Bus +5/+12/-12 VDC power supply failure) in the Unit 2 LEFM. The Unit 1 LEFM was different in that it had redundant power supplies and was not vulnerable to this failure.

PALO VERDE STATION

On August 10, 2002, at 1500 hours, Palo Verde Unit 1 received Feedwater UFM (Ultrasonic Flow Measurement System) related alarms.

Representatives from the vendor (Caldon) replaced two computer boards, and adjusted the coarse gains and peak detect signals. Caldon verified the LEFM was performing within its ± 1.0% uncertainty value.

The unit down power resulted in an estimated generation loss of approximately 2600 MWhr, which contributed an approximate 0.26% Unplanned Capability Loss Factor (UPCLF) for the month of August.

GRAND GULF STATION

The presence of a "Low Virtual Memory" alarm on the LEFM panel screen was thought to be a precursor to the system going off line. On November 27, 2002, a "Low Virtual Memory" alarm was generated on the front panel screen causing the Caldon LEFM Check Plus System to be lost.

The vendor was contacted and advised Grand Gulf personnel to acknowledge the alarm and allow the system to continue to operate. The vendor initiated an internal deficiency document and started to review the operating system and software for potential problems. Approximately two weeks later, the system unexpectedly went off-line for approximately three minutes. When the system returned itself to service all parameters appeared to be normal. Caldon was notified of this occurrence and another deficiency document was generated and research efforts were increased. At that time, it was not certain if the problem resided within the software or the mainframe of the LEFM.

Later, Caldon was able to duplicate the problem as two more plants reported the same problem and was able to make changes to prevent it from re-occurring.

PEACH BOTTOM STATION

On December 12, 2002, with Unit 2 at 100% power and producing greater than expected electrical output, operability of the LEFM system could not be confirmed and thus, per TRM 3.20, reactor power was lowered from 3514 MWth to 3458 MWth.

The cause was determined to be low-impedance grounds that affected the accuracy of the LEFM feedwater flow measurement.

A later load drop on February 21, 2003, to repair some leaks, was extended by 12 hours when repairs to the leaks caused several transducers to fail.

RIVER BEND STATION

On May 10, 2003 it was discovered that the Caldon model 8300 strap-on LEFM in use at the station was giving a non-conservative feedwater flow that caused the plant to operate at 102.7% power for over 15 months. According to the manufacturer, the analyses of the LEFM data, as well as data from the feedwater flow venturis and the feedwater RTDs, have established that there were three contributing causes of the LEFM's errors:

1) There were non-conservative assumptions made regarding calibration of the instrument,

2) Changes that occurred when feedwater flow was significantly increased due to a 5 percent power uprate produced non-conservative feedwater flow indication error, and

3) A measurement error during installation resulted in non-conservative error in the indicated flow reading. Therefore, when the external LEFM 8300 data was used to correct feed flow for core thermal power calculations, the LEFM 8300 was not providing data within its specified accuracy.

The LEFM was removed at a later outage and a new program to calculate a best estimate for reactor power was developed.

POINT BEACH STATION

Reactor thermal output was observed shifting from the LEFM measurement system to the feedwater venturi. Operators responded conservatively by reducing power by approximately 2% to account for potential uncertainties with the feedwater venturis. The cause for the equipment shift was due to a faulty hard drive card that was replaced.

COFRENTES STATION

An anomalous response of the LEFM was observed at Cofrentes NPP during the startup following the plant outage of October 2011.

The root cause was determined to be debris caught in the flow conditioner in feedwater line A. which altered the axial velocity profile seen by the Loop A LEFM. The altered profile apparently introduced bias in the Loop A LEFM calibration.

While this was not a failure of the LEFM itself, it highlights the ease with which any change in hydraulic parameters can change the flow profile the LEFM depends upon for calculation of flow rate.

LIMERICK STATION

On January 27,2012, the Limerick Unit 1 LEFM system entered MAINTENANCE mode due to a hydraulic asymmetry alarm.

There were no corrective actions as the cause was determined to be changes in the flow profile based upon the taking the #6 FW Heaters out of service.

HOPE CREEK STATION

On June 28, 2002 a detailed analysis of the event on May 26, 2002 determined that ultrasonic flow detection equipment used to detect feedwater flow was malfunctioning because of calcium silicate insulation lodged between the clamp and pipe. The malfunction of the cross-correlation instrumentation correction factor resulted in the reactor exceeding its licensed power level by an average of 0.25% for eight hours.

Immediate corrective actions included removing the cross-correlation instrument from service and reducing power below the 1.4% uprate value. Additional corrective actions included the installation of new cross-correlation transducers and validation of the existing cross-correlation performance. The mounting configuration was also changed.

1.5 Organization of Topical Report

The Topical Report will be organized in the following sections:

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Brief descriptions of the various project tasks associated with the development of the Topical Report are discussed in the following sections.

2.0 Licensing and Basis Requirements

The feedwater flow rate measurement devices (venturis, nozzles, ultrasonic flow meters and associated instrumentation) used at BWRs and PWRs for the determination of licensed reactor core power are typically supplied and installed to high quality, commercial standards.

Feedwater flow measurements systems are typically classified as non-safety related components and systems. The plant's Updated Final or Final Safety Analysis Report (FSAR) contains the safety classification of the feedwater flow measurement system.

Feedwater flow measurements systems may be classified as important measurements that are used during and following an accident.

Results from the DVR calculations of plant measured data may be used to implement a recalibration of the licensed core power feedwater flow rate measurement devices or "Power Recovery". In this application, the DVR method's accuracy or uncertainty must be sufficient to ensure the site's licensed reactor core power uncertainty, typically 2%, is maintained.

In 2000, the Nuclear Regulatory Commission (NRC) amended 10 CFR 50, Appendix K, to provide licensees the option of maintaining a 2% power margin or applying a reduced margin based on the improved measurement uncertainties resulting from incorporation of more accurate feedwater flow measurement instrumentation.

Several Appendix K uprates (MURs) have been performed in the U.S. using higher accuracy ultrasonic feedwater flow metering systems. The process required an NRC approved license amendment as the safety margins for the plant are affected.

One potential application of the DVR process is to use the calculation results to facilitate a Measurement Uncertainty Recapture (or MUR).

2.1 Safety Classification

The safety classification for DVR technology may depend on the application. For MURs where licensed margins could be affected, additional Quality Assurance measures may be required for program implementation.

2.2 Discussion of the application of 10CRF Part 50, Appendix B to the design and implementation of the DVR model

2.3 Applicable Codes & Licensing Requirements

2.4 Software Quality Assurance and Verification and Validation (V&V Process).

2.5 Cyber security.

2.6 Regulatory Compliance – Table showing reference, description of guidance and section of topical where Guidance is addressed.

3.0 Methodology

3.1 Theory of Data Validation & Reconciliation (DVR)

3.1.1 Statistical basis for uncertainty

By referring to the uncertainties of the DVR result variables and the assumption of normal distribution justified by the central limit theorem and by employing the distribution function of the normal distribution, it is therefore possible to state the statistical certainty (probability) with DVR calculation of CTP with a Confidence Interval of 95%.

3.1.2 Applicability with ASME PTC Codes

ASME 19.1 Section A.2 describes a general methodology for determining a weighted overall uncertainty whenever a value of a parameter is approximated by several different measuring methods. However, the ASME codes do not address use of empirical covariance matrices and Gaussian corrections as a means determining instrumentation and test uncertainties. Therefore, at this time, VDI-2048 is the primary source for the engineering and statistical calculations as related to the DVR process. As part of this project, applicability of the DVR method to the ASME codes will be investigated. The addition of an appendix to ASME 19.1 for DVR is under consideration by the committee.

3.1.2.1 19.1 Appendix A: Statistical Considerations

3.1.2.2 19.1 Appendix C: Propagation of Uncertainty Through Taylor Series

3.1.2.3 19.1 Appendix D: Central Limit Theorem

3.2 Comparison with International Codes (VDI-2048)

3.3 Mathematical Basis for DVR

The Topical Report will provide a detailed mathematical explanation of the Data Validation and Reconciliation process. Below is a summary of the process.

The measurement values are improved by means of a correction calculation, which applies conditions based on mass flow balances and energy balances. In the case where flow is split between two pipes with three flow measurements, the relationship between the measured variables exists which will never quite fulfill the physical law of conservation of mass.

Functional redundancy results from applying the physical properties and relationships between various measurements. Figure 1 is a simple example of a functional relationship between parameters.

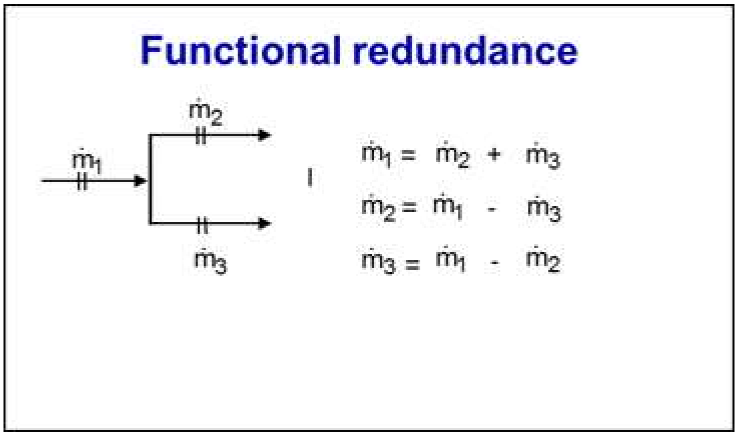

Figure 1

The DVR process introduces the concept of using an Empirical Covariance Matrix to analyze the plant measured instrumentation for errors. The matrix is the covariance of the measurement as related to the variance, where the variance is the square of the normal distribution.

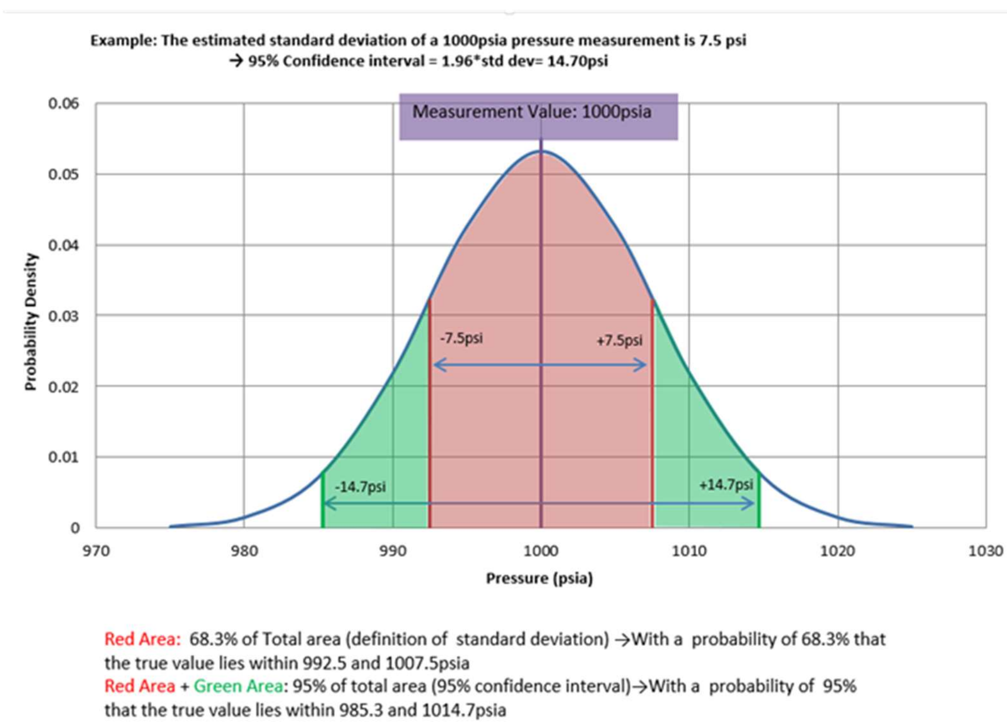

Figure 2

Normal or Gaussian Distribution – Two Standard Deviations (1.96) is a 95% Confidence Interval.
Variance is the Standard Deviation Squared.

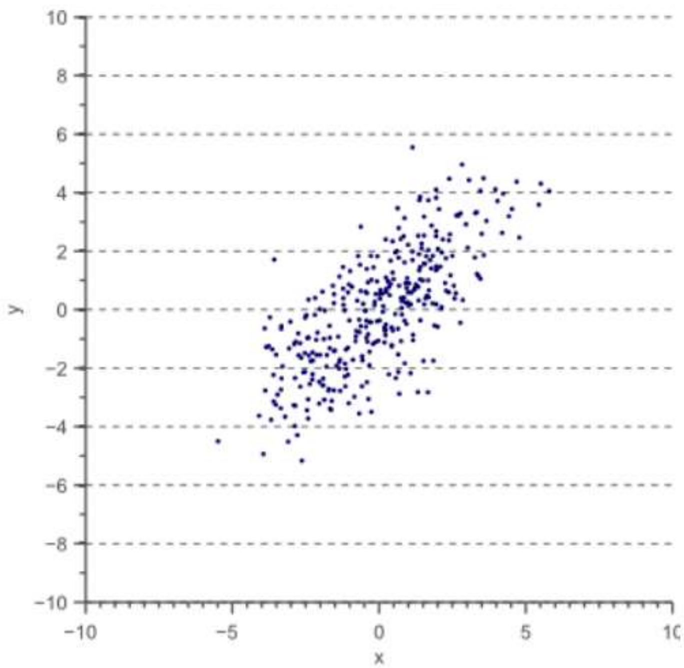

Figure 3

Graphical Representation of Covariance Matrix. X and Y axes represent Covariance of the Measured Data

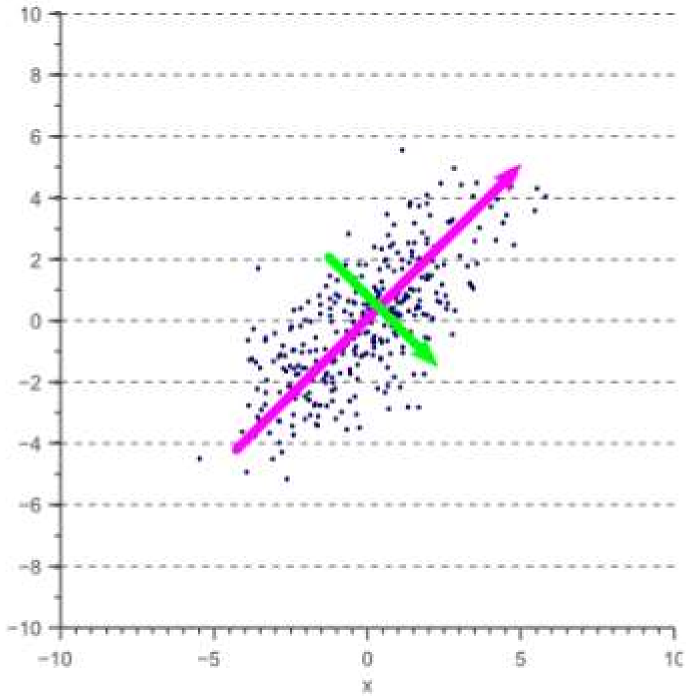

Figure 4

Linear transform of the Covariance Matrix for largest variance of the data (subject to the "auxiliary conditions").

Process and quality control functions are applied to validate the results of the overall evaluation as well as individual measurements.

- Estimates from the DVR calculation are subject to the mass-energy and other engineering calculations specified in the DVR model. These are referred to as "auxiliary conditions".

- **Reconciliation** (closing all the balances with the Gaussian correction principle)

Objective Function = $\sum \left\{\frac{measured\ value - reconci\quad value}{standard\ deviation}\right\}^2 \rightarrow minimum$

The objective function must be less than a Chi-Square test to have assurance that the corrected values are acceptable. The Chi-Square test is a “goodness of fit” test to ensure all measurements are reconciled within 2 standard deviations difference (95% confidence).

- **Quality control for the whole process**

$$\text{Quality} = \frac{Objective\ Function}{95\%\ quantile\ of\ Chi^2} < 1$$

- **Quality control for each measurement**

$$\text{Single Penalty} = \frac{(measured\ value - reconcil\quad value)^2}{\max(correction\ unc.^2, \frac{measurement\ unc.^2}{10})} * (1.96^2) \leq 3.84$$

Measurements with Single Penalty values equal to or exceeding 3.84 (95% confidence interval in terms of variance) are considered suspect.

Included in the evaluation of the Mathematical basis will be a detailed explanation of the following:

3.3.1 Derivation of redundant physical relationships.

3.3.2 Demonstration of the use of empirical covariance matrix, propagation of errors, and Gaussian correction method.

3.3.3 Demonstrate the determination of calculated Heat Balance results against an independent calculational methodology. Using DVR results as input to a thermodynamic model comparing key performance parameters such as Core Thermal Power and Feedwater Flow.

3.3.4 Identify the most significant Core Thermal Power error contributors. Use an independent thermodynamic model of the unit to conduct sensitivity studies for those variables to develop a simplified correlation matrix to compare to an equivalent matrix used in the proposed solution methodology.

3.3.5 Perform sensitivity analysis using DVR software.

3.4 Empirical Comparisons

3.4.1 Demonstrate DVR accurately predicts the calibration state for a number of instruments. This would also establish lab traceability to the DVR method.

3.4.2 Compare DVR results to an operable feedwater flow instrument of known accuracy.

3.4.3 Demonstrate DVR capability to identify plant events which cause reduced quality of the results.

3.4.4 Demonstrate the repeatability/reliability of DVR results by comparison with plant data over a period of time.

4.0 Benchmarking and Operational Experience (OE)

The benchmarking will consist of showing actual applications of DVR as compared to other plant flow metering and core thermal power calculational methods. It will also provide empirical evidence of the reliability, repeatability and robustness of the process. This section will also include examples of the process as it is currently used in the world wide nuclear power industry.

5.0 Application of DVR Based on Power Plant Data

5.1 Developing generic methodology

5.1.1 Describe process of extracting Design data from Heat Balance diagrams, turbine vendor thermal kits, software heat balance models (if available), plant calorimetric calculations, heat exchanger data sheets, pump data sheets, and any other plant design data that may be used in the model.

5.1.2 Use of plant piping and instrumentation drawings and circuit wiring diagrams to determine the DVR configuration and instrument locations in the model.

5.2 Determining instrument inputs

5.2.1 Certain plant measurements are more correlated to the CTP than others. The CTP correlation factors may be determined by the use heat balance and DVR model sensitivity studies. These studies will be performed to identify the most important plant measurements.

5.3 Determining instrument calibration requirements

5.3.1 Certain plant measurements are more correlated to the CTP than others. As a result, the relative accuracies of these measurements may be more important to maintaining accurate and repeatable DVR CTP calculations. The accuracy requirements for these instruments will be examined.

5.4 Justification of the use of multiple instruments with lesser accuracy

5.4.1 In general, the accuracy of a plant measurement can be improved through the use of multiple instruments, provided the local conditions measured are similar. Analytical studies will be performed to determine plant measurement locations that may be measured with multiple, less accurate instruments instead of one highly accurate measurement.

5.5 Determining software settings

5.5.1 Estimating the accuracy of method

5.5.2 Checking the methodology against real or synthetic plant data to ensure the selected method is viable for its intended use.

5.6 Demonstrating the approach for use in error detection of UFM or venturi (or nozzle) indication.

5.7 Confirmation that the impact of the deficiencies inherent in the on-line monitoring technique have been quantitatively bounded and accounted for either in the on-line monitoring acceptance criteria, and the impact on plant safety will be insignificant. The deficiencies may include inaccuracies in process parameter estimate single-point monitoring and traceability of those accuracies to known standards.

5.8 Description of bounding conditions for use of results.

5.9 Demonstration that the DVR process can distinguish between the process variable drift (actual process going up or down) and the instrument drift.

5.10 Demonstrate that the DVR process can compensate for uncertainties introduced by unstable processes, sensor locations, non-simultaneous measurements, and noisy signals. If the implemented algorithm-and/or its associated software cannot meet these requirements, provide a description of administrative controls put in place acceptable means to ensure safe application of the results.

5.11 Clear demonstration of the relative importance of measurements with respect to the determination of core thermal power.

5.11.1 Description of internal algorithms which are used to determine if these measurements are acceptable.

5.11.2 Description of Mitigation techniques to recover acceptable results in the event of a failure of an instrument important to the overall results.

5.11.3 Identification of methods to reduce the overall impact of an instrument failure (e.g. addition of a redundant instrument).

5.11.4 Demonstration that immeasurable modeling assumptions are bounded by the DVR uncertainty.

5.11.5 Demonstration of the removal of any Bias in the results.

5.11.6 Demonstration of how changes in the process such as steam leaks or component failure will be captured by the process quality controls.

5.12 Demonstrate satisfaction of the following:

5.12.1 Identify limitations of the application of the DVR results based on non-steady state operations.

5.12.2 Estimation of variance-covariance structure for the measured variables

5.12.3 Determination of the underlying probability distributions of the measurements.

5.12.4 Application of Redundancy criteria- the measurement can be uniquely calculated from a set of the measured variables that are consistent with the constraints.

5.13 Description of DVR software attributes

5.14 Discussion of plant data interface.

6.0 Demonstration Application

6.1 Description of administrative controls for implementing results to power calculation

6.2 Description of inputs to software

6.3 Description of Design Control and Documentation Requirements

6.3.1 Design Specification

6.3.2 Software Controls

6.3.3 Functional Acceptance Test

6.3.4 Site Acceptance Test

7.0 DVR Failure Modes Evaluation

Objectively determine the extent to which DVR methods can provide quantifiably, repeatable, accurate results that may be used as a licensed power backup or replacement determination [Formal Calculation provided[2]].

[2] Formal Calculation to be developed for the Topical Report.

Analytical studies will be performed to identify DVR models areas of weakness. These studies will include failure or inaccurate plant measurements used as inputs to the DVR model. Incorrect assumptions of design data used as DVR inputs will also be considered. Unmodeled conditions of the model, such as excessive valve leakage or unknown plant component degradations or failures will be evaluated. From these studies a set of bounding CTP uncertainties and rules for times when the DVR model may not be usable will be developed.

Data representation may also be subjected to affinity diagraming to identify potential causes of failures in the process into groupings showing areas that should be focused on the most. Cause and effect diagrams such as fishbone, Ishikawa diagrams will be used to help identify potential causes of failures within the process. Histograms could be utilized to show processes that are non-compliant or represent failures in the process. Flowcharts, matrix and scatter diagrams may also be utilized to show data representations. Matrix diagrams if utilized will show strength of relationships among variables. Pareto charts may also be used to illustrate potential cumulative effects to aid in focusing attention of process causes that could lead to defects or failures of the technology.

7.1 Failure mechanisms of the technology [Formal Calculation][3]

7.2 Failure modes

7.2.1 Process for Self-identification of software failure mechanisms

7.2.2 Instrument failures

7.2.3 Component (plant equipment) failures

7.2.4 Calculational error identification

7.3 Determination of Uncertainty

Evaluate and establish the objective criteria by which the results may be used Identify the instrumentation requirements: number and quality of instruments, necessity for calibrations, etc. Perform formal sensitivity analysis to identify areas of bias [Formal Calculation[3]].

7.4 Application of Uncertainty

The basis for objective criteria by which it can be determined that the technology is meeting its intended purpose [Formal Calculation[3]]

7.5 Limitations

7.5.1 Quality Assurance procedures for the technology (hardware and software) Nuclear grade Quality Assurance practices will be implemented for this phase of the project.

[3] Formal Calculations to be developed for the Topical Report

8.0 Off Normal Operations

The results from DVR will be evaluated during Off Normal Operations. Off Normal Operations will be those times of plant operation when system line-ups are configured differently than typical normal operation and/or when a component or system is operating with degraded performance. This data will be used to help determine limitations of DVR.

9.0 DVR Reliability

The reliability of DVR for installations currently in use will be assessed. In addition, sensitivity studies and evaluations of plant data will be performed to identify DVR model inputs and plant measurements that could affect the reliability of the DVR results. Fix-up or corrective routines that can be implemented in the software set up or handled administratively, will also be evaluated. This will aide in:

9.1 Prevention of Non-Conservative results

9.2 Robustness of calculation due to diversity of inputs

10.0 LCO Conditions

Sensitivity studies and evaluations of plant data will be performed to identify conditions when the DVR performance may be affected such that the results exceed the expected accuracy, or the DVR calculations fail. LCOs, the section of Technical Specifications that identifies the lowest functional capability or performance level of equipment required for safe operation of the facility, apply to the performance of the plant calorimetric (CTP) at some plants. The potential to affect LCO conditions through the use of DVR will be evaluated.

10.1 International applications

10.2 Comparison to similar applications

11.0 Limitations

Determining the acceptable time durations for when an alternate or backup calorimetric may be used. Determining plant operating conditions that would limit the application of the DVR correction factor.

12.0 Conclusion

A Conclusions section will be prepared.

13.0 References

References will be provided.

14.0 Appendices

Appendices will be provided as needed.