WESTINGHOUSE PROPRIETARY CLASS 3

WCAP-10967

TOPICAL REPORT

EVALUATION OF THE WESTINGHOUSE BWR CORE MONITORING SYSTEM

September, 1985

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ABSTRACT

An evaluation of the core on-line monitoring software system used by Westinghouse for boiling water reactor applications has been performed based on an uncertainty analysis of the calculational model and instrumentation measurement system. The calculational model and instrumentation uncertainties are presented and discussed. The propagation and synthesis of these uncertainties into the on-line evaluation of the core thermal limit parameters is performed using a Monte-Carlo error simulation. The results are presented as percent standard deviation of the evaluated power distribution and core thermal limit parameters.

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1.0 INTRODUCTION

The objective of this report is to document the evaluation of the COMPUS (Core On-line Monitoring POLCA and UPDAT Software) computer software system intended by Westinghouse for use in performing the on-line core monitoring function of boiling water reactor (BWR) cores. This report contains a description of the measured process parameters used and their associated measurement uncertainty, a description of the computer calculational modules, and a description and presentation of the results of the Monte-Carlo error analysis performed to evaluate the overall uncertainty of using this system to monitor the reactor core.

The COMPUS system is a software system developed for performing on-line monitoring of operating conditions in BWR cores at U.S. utility installations. The COMPUS system is an extension of the POLCA/UPDAT system developed by ASEA-Atom and has been used for core monitoring of some 70 reactor cycles of operation at BWR installations in Sweden and Finland. The system provides an efficient and reliable computer software package for monitoring the core operating margins to insure the structural integrity of the fuel. The thermal limit parameters monitored are the linear heat generation rate (LHGR), average planar linear heat generation rate (APLHGR) and the critical power ratio (CPR).

The COMPUS system is a minicomputer based, software system designed for installation at the utilities' reactor site. The minicomputer where COMPUS resides is data-linked to the present plant process computer used for data acquisition of the measured core parameters. The CONDIN module of the COMPUS system is responsible for accessing the on-line operating data from the process computer. CONDIN accesses the process computer where the measured process parameters are stored and makes available to COMPUS the process parameters for use in the evaluation of the current operating conditions.

The COMPUS system utilizes both the measured process data and the neutron flux measurements from the in-core Local Power Range Monitors (LPRMs) and the Traversing In-Core Probes (TIPs) for determining the total reactor power and evaluating the core power distribution. The limited number of detector readings and the possible existence of localized power peaking necessities a

model that determines the power in regions away from the detectors. Such a model is provided for in COMPUS by using a three-dimensional simulation of the reactor core at the operating conditions. The actual operating conditions (i.e., control rod pattern, core power, core flow, exposure, xenon concentration, etc.) and up-to-date core nuclear data are used in establishing this model of the reactor core. This simulation is performed in the POLCA module of COMPUS and provides a realistic nodal simulation of the conditions in the reactor core to determine the power distribution at the time.

The actual determination of the thermal limit parameters begins with this "base" calculated distribution of nodal powers throughout the core. This power distribution is adjusted by an interpolation function multiplier derived by appropriately weighting the differences between calculated and measured detector readings of the nearest detectors. The modified power distribution, properly normalized, is then utilized to determine the operating thermal limit parameters in terms of LHGR, APLHGR, and CPR. The step of determining the modified power distribution and the actual operating thermal limit parameters is performed in the UPDAT module of COMPUS.

The performance of the COMPUS system has been evaluated based on a Monte-Carlo analysis of the measurement and calculational systems. This report identifies the significant sources of uncertainties in the COMPUS system and determines the effects of these uncertainties on the core power distribution and thermal limit parameters of the reactor core. The results consist of the standard deviations of the nodal power distribution and thermal limit parameters for a chosen set of core locations and plant conditions during normal steady state operation. The errors include instrumentation, calculational models, and manufacturing uncertainty but do not consider gross failures of the measurement systems. This evaluation was performed on an off-line version of the system.

2.0 CONDIN Module

The COMPUS core monitoring software system is designed to reside on a minicomputer which is data-linked to the present plant process computer used for data acquisition of the measured core parameters. The CONDIN module of the COMPUS system will periodically access the on-line operating data from the process computer and make this data available to the POLCA and UPDAT modules of the system.

The on-line data required by the POLCA and UPDAT modules includes the reactor core thermal power level, the reactor core coolant flow rate and temperature, the reactor operating pressure, the plant gross electrical power, the control rod position indicator readings and the neutron detector TIP and LPRM readings.

It is assumed that various modules of the existing process computer remain available to provide the data acquisition function required to interface COMPUS with the reactor core instrumentation. In particular, the modules which perform the function of data acquisition of the TIP and LPRM measurements, control rod settings, and the other measured process parameters are assumed to be in place. Also required are the modules responsible for the data validity and reading substitution for the TIP readings, LPRM values, and control rod position indicators. In addition, the module used to determine the reactor core thermal power level is required to be in place.

The measurement uncertainties of these process parameters in the evaluation of the total uncertainty associated with the COMPUS core monitoring function are the uncertainties associated with the measured reactor core power level and the TIP and LPRM measurement uncertainties. The measurement uncertainties used in evaluating the COMPUS uncertainty are those reported in Reference 3. The uncertainty in the measured reactor core thermal power level was taken as 1.76% while the uncertainty in the measured TIP detector readings was taken as 2.6% and the LPRM detector reading uncertainty as 3.4%.

3.0 POLCA MODULE

The objective of the POLCA module in the COMPUS system is to provide a three-dimensional model of the core to evaluate the local conditions in the core with a high degree of accuracy. In order to achieve this accuracy POLCA realistically simulates the nuclear and thermal-hydraulic behavior in the BWR reactor. This modified one-group nodal code is used to calculate a core power and void distribution for subsequent use in the UPDAT module for evaluating the thermal limit parameters LHGR, APLHGR, and CPR.

The POLCA module is used to evaluate the core power and void distribution following any core operating change which results in a localized change (i.e., control rod movement). It is also used to evaluate the core conditions periodically to incorporate the effects of the operating history over that time interval. By providing this detailed model of the core the accuracy of core tracking (i.e., exposure accumulation) is greatly enhanced and results in improved core monitoring evaluations throughout the operating cycle.

In the following sections a brief description of the models employed in POLCA is given. A more comprehensive description is available in Reference 1. This POLCA module is calculationally identical to the POLCA code benchmarked in Reference 2.

3.1 Neutronic Model

In POLCA the three dimensional reactor core is divided into nodes in which neutronic characteristics are described by homogenized two group macroscopic cross-sections determined for each fuel assembly type in the core using a pin cell type code such as PHOENIX (also described in Reference 1). The neutron transport between nodes is calculated using a one-group, coarse mesh diffusion theory method which employs an albedo treatment to represent the core-reflector interface. A thermal flux correction is used to simulate the thermal neutron transport providing a flux solution that closely approximates that of a rigorous two group calculation. The three dimensional power distribution calculated in POLCA includes the effects of the coolant flow and

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void distribution, the presence of control rods, as well as, reactivity effects due to fuel temperature Doppler effects and the xenon concentration in the core.

3.2 Core Geometry

In POLCA the fuel region of the core is divided into a large number of equal sized rectangular parallelepiped zones or nodes. Within each node the fuel composition, exposure, void content and power density are constant. However, these nodal parameters do vary from one node to another. Each node in the core is identified by a fuel type to describe its characteristics. In each radial plane the nodes are square and generally correspond to a fuel assembly's width, whereas each fuel assembly in the axial direction is divided into at most 25 axial nodes. This node division produces approximately cubic nodes with sides of approximately 6 inches in width. The neutron transport in POLCA is limited to the six nearest neighboring nodes surrounding that node.

In addition to the capability of modelling the full core geometry in three dimensions, POLCA is capable of taking advantage of radial core symmetry through appropriate boundary conditions. The module is capable of handling half and quarter core calculations assuming either reflective or rotational symmetry.

The characteristics of the core-reflector interface, both in the radial and axial direction are described in POLCA using an albedo treatment to indicate the probability that a neutron leaving the core and reaching the reflector will be reflected back into the core.

3.3 Cross Section Treatment

Homogenized two-group cross sections, obtained from a pin cell type code, such as PHOENIX, are input to POLCA as a number of 3-dimensional tables with a table for each fuel type. The tables are functions of exposure (E), void content (V) and conversion history (C). The conversion history is defined as the sum of the void history (exposure weighted average void fraction) and the

control rod history (exposure averaged control rod presence converted to a void fraction).

The macroscopic cross-section parameters required by POLCA include D_1, D_2 Σ_{rem}, Σ_{a1}, Σ_{a2}, Σ_{f1}, Σ_{f2}, and ν, where all parameters have the conventional definition.

The effect of the control rods is incorporated through the absorption cross sections Σ_{a1} and Σ_{a2}. If a control rod covers a node, the absorption cross-sections are modified by the $\Delta\Sigma_{a1}$, and $\Delta\Sigma_{a2}$ terms to account for the control rod presence.

3.4 Neutronic Feedback Model

The homogenized two group cross-section parameters supplied to POLCA correspond to reference conditions used in the lattice cell calculation. POLCA therefore must modify the cross-section data to account for core conditions differing from the reference conditions. The two neutronic feedback effects modelled in POLCA are the Doppler fuel temperature and xenon concentration. POLCA accounts for fuel temperature differences from the reference temperature through modifications to k_∞. Xenon concentrations differing from equilibrium concentrations are accounted for through modification of the thermal macroscopic absorption cross-section, Σ_{a2}.

3.5 Power Distribution Calculation

The three dimensional neutronics of the reactor core is modelled in POLCA by an effective one group nodal model. Coupling coefficients describing the inter-nodal coupling are evaluated from the two group data and take into account the local spectrum mismatch effects. The neutronic coupling in the reactor core is assumed to be primarily due to the fast neutrons which have a relatively large mean free path. POLCA, however, does modify the flux solution to account for thermal neutron migration and thus obtains solutions in good agreement with rigorous two-group diffusion theory. From the converged neutron flux distribution the three-dimensional nodal power distribution is determined for the core.

3.6 Thermal-Hydraulic Model

In the POLCA hydraulics model the total coolant flow entering the core is known. Computing the flow through each fuel assembly (channel) is the primary objective of the hydraulics calculation. The three dimensional model for the reactor core is based on a nodal mesh description the same as that used in the neutronics model. In both the radial and axial direction each node corresponds to the nodal geometry used in the neutronics model.

The calculation begins at the core inlet and continues, by node, from the inlet to outlet node for each fuel assembly channel. The pressure drop across each node is evaluated, and when integrated over the entire channel, yields the total pressure drop for the assembly. Having obtained the total pressure drop for each assembly, the core average pressure drop is computed, and the assemblywise flow distribution is evaluated. The code iterates on the flow distribution through each fuel assembly until the pressure drop calculated across each assembly is equal to the core pressure drop.

Once a converged flow distribution has been obtained, the three-dimensional void distribution is calculated using the void correlation described in Reference 4. This void distribution is coupled with the power distribution in the power-void loop in POLCA.

3.7 Power-Void Loop

In a BWR core simulator program the thermal hydraulics model has a significant effect on the calculated core reactivity and power distribution. The neutronic and thermal hydraulic models in POLCA are coupled through the power and void interaction encompassed in the power-void iteration loop of the code. This iteration uses the three-dimensional power distribution from the neutronics calculation directly in the evaluation of the three-dimensional void distribution. An iterative procedure is utilized until consistent power and void distributions are obtained from the calculations.

3.8 Detector Model

The LPRM and TIP neutron flux detector readings, as obtained from the process computer, yield information on the actual state of the core. The detector readings provide the most important and the most accessible information concerning the power distribution in the reactor core. In order to couple the results of POLCA to the measurements from the reactor, POLCA is required to simulate the LPRM and TIP detector readings.

The detector readings are proportional to the neutron flux at the detector location. However, because POLCA does not involve sufficient details about the neutron flux at the detector locations, the detector readings must be calculated from the power distribution in the surrounding nodes of the detector. The detector readings are considered proportional to the power in the adjacent nodes of the horizontal plane of the detector. The absolute reading is not of utmost importance due to the intercalibration of the detectors.

The constants of proportionally between the nodal powers and the detector readings, K_i are determined by PHOENIX and supplied to POLCA as a function of fuel type, exposure, void, and void history. There also is a very strong dependency to control rods inserted in the vicinity of the detectors. The control rods effect depends on the distance of the control rod to the detector, and for this reason, correction factors, KSS_i are used. The exact position of the detector is represented in PHOENIX to generate the K_i and KSS_i tables.

The detector readings are calculated in POLCA as:

$$D = \frac{1}{N} \sum_{i=1}^{N} \frac{P_i}{K_i \cdot KSS_i} \qquad (3.8\text{-}1)$$

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Where P_i is the nodal power of N surrounding nodes horizontally adjacent to the detector. The detector readings are determined for both the Traversing In-Core Probes (TIP) and the Local Power Range Monitors (LPRM).

3.9 Exposure Updating

In order to provide a valid core model throughout the operating cycle, POLCA must be capable of performing exposure updating of the core. In an exposure update, new values are calculated for exposure (E), void history (VH = exposure void), and control rod history (SH). The sum of the void and control rod histories is referred to as "conversion history." It is assumed that during the exposure update the xenon is in equilibrium with the power, and that the power distribution is constant during the exposure step.

The void history is intended to take into account the fact that the isotope structure of the fuel (especially the plutonium content) is dependent on the void content during the earlier phases of the burnup process. The void history is accounted for by an exposure-averaging of the instantaneous void.

Due to the influence of the control rods on the neutron spectrum, the isotope content will be affected in the sections of the core where control rods have been inserted. The impact is very similar to that obtained from an increased void content. Therefore, control rod history is treated in POLCA as an equivalent amount of void history.

Updating of control rod history takes place independent of the void history calculation. However, since the impact of these two magnitudes on the fuel is of the same nature, they are added together to give the conversion history before interpolation in the cross section tables.

3.10 XENON TRANSIENT MODEL

During a POLCA calculation, xenon and iodine can either be in equilibrium with the power distribution, or they can be in a transient phase as a result of a power disturbance.

The differential equations that describe the contents of I-135 and Xe-135 can be solved analytically, if it is assumed that the power in the node is constant during time step Δt. Since the xenon feed-back on the power is a magnitude smaller than the effect of the power on the xenon content, it is possible to take relatively long time steps (normally several hours) during a transient calculation.

At the beginning of each time step, a void/power distribution is calculated which is consistent with the specified xenon distribution. Based on this power distribution, the xenon distribution is then updated with the time step, Δt, and a new void/power distribution is calculated. Then, before the next time step, the reactivity is checked and/or the coolant flow or the reactor power is tested to see if a change is required.

4.0 UPDAT MODULE

The UPDAT module of the COMPUS system is where the evaluation of the core thermal limit parameters takes place. The basis for the UPDAT evaluation are the results from the "base" POLCA evaluation and the most current measured process values and measured neutron detector readings. The UPDAT module adjusts the POLCA power profile such that the detector readings which could be inferred from this modified power profile would agree with the actual measured detector readings. Adjusting the power profile in this manner forces the calculated (POLCA) profile to agree with measurements, thus producing an inferred "measured" power profile of the core. From this measured power profile the operating margins of the core are determined. The thermal limit parameters determined by UPDAT are those concerned with the linear heat generation rate (LHGR), average planar linear heat generation rate (APLHGR) and the critical power ratio (CPR). An example would be the variations which occur in the power and flow rates during normal operation of the plant. It is anticipated that the UPDAT evaluation will occur on a relatively frequent time interval generally on the order of every several minutes of core operation.

It is important to keep in mind that the objective of UPDAT is to correct the POLCA calculated power profile for those variations in core operating conditions which effect the core on a global basis.

In the following sections is a description of the calculational models employed in UPDAT which this report addresses in the way of accuracy and uncertainty.

4.1 Power Distribution Evaluation

The first step in evaluating the core thermal limit parameters is to determine the three-dimensional nodal power distribution of the core at its current operating condition. This nodal power distribution is determined taking into account the most current measured LPRM readings. With these measured LPRM readings UPDAT corrects the latest "base" power distribution calculated by the POLCA module to account for the differences in the current core operating

condition at time t_{core} to the operating conditions of the core at the time of the POLCA calculation (t_{POLCA}). This correction is accomplished using the measured LPRM readings and the LPRM readings predicted by POLCA to correspond to the calculated power distribution.

Due to the limited number of LPRM measurements available in the axial direction UPDAT compensates by using the latest available TIP data to improve the data density in the axial direction. At the time of the TIP calibration (t_{TIP}) a basic TIP correction factor is calculated for each measured TIP distribution. This TIP correction factor is calculated as the ratio of the measured TIP readings to the corresponding POLCA calculated readings. Thus for each TIP detector location a correction factor is computed for each axial position as:

$$TC_i = TIP_i^{measured} \; / \; TIP_i^{POLCA} \tag{4.1-1}$$

The measured LPRM data obtained from the process computer is used to adjust this TIP correction factor, TC, with respect to changes in the core each time (t_{CORE}) an UPDAT evaluation is performed. Please note, in the COMPUS operational sequence $t_{TIP} \leq t_{POLCA} \leq t_{CORE}$ (see Figure 4-1). The TIP correction factor is adjusted by first computing the ratio of the measured LPRM readings and POLCA calculated LPRM readings. This ratio is then combined with the TIP correction factor to determine an L-factor for each of the four LPRM detectors in the string at the location of the TIP measurement. The L-factors are computed as:

$$L_j = LPRM_j^{measured} \; / \; LPRM_j^{POLCA} \; / \; TC_j \tag{4.1-2}$$

Finally a modified TIP correction factor is computed as:

$$T_i = TC_i \; ((1-W_i) \cdot L_j + W_i \cdot L_{j+1}) \tag{4.1-3}$$

for each axial node, which provides an axial correction factor for the power distribution over the entire axial direction of the core. The W_i term is an axial geometric weighting factor to linearly correct for nodes between the

LPRM positions, and the L_j and L_{j+1} are the L-factors for the nearest LPRMs above and below axial node i.

Notice that at the LPRM levels the above equation reduces to:

$$T_j = LPRM_j^{measured} / LPRM_j^{POLCA} \qquad (4.1\text{-}4)$$

Figure 4-2 illustrates how the axial TIP correction factor is modified based on the most current LPRM readings for each TIP detector location in the core.

After UPDAT has determined the modified TIP correction factor at each of the TIP detector locations the UPDAT module proceeds to correct the POLCA calculated three-dimensional nodal power distribution using these correction factors. Since only approximately a quarter of the fuel assemblies in the core are located with a neutron detector instrumentation thimble adjacent to them UPDAT must therefore use an interpolation/extrapolation algorithm to correct the assembly power which are located away from an instrumentation thimble. Figure 4-3 illustrates the LPRM and TIP location on the radial plane.

Knowing the radial position of each detector, UPDAT will find the closest detectors which surround the assembly whose power distribution is being corrected. It should be mentioned that the UPDAT calculation is being performed for the entire full three-dimensional core, meaning there are no presumed symmetries in the core and that the measured detector data is not being used to correct assembly powers other than those which physically are located near them.

Once UPDAT has determined the surrounding detectors for the assembly of interest UPDAT adjusts the POLCA calculated power according to:

$$P_i^{UPDAT} = P_i^{POLCA} \cdot \sum_{k=1}^{K} W_k \cdot T_i^K \qquad (4.1\text{-}5)$$

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where the weights W_k when summed over the surrounding detectors are equal to 1.0. The summation is over all surrounding detectors. In the interior of the core K is equal to 4, and at the core periphery K is equal to something less than 4. This weighting function is illustrated in Figure 4-4.

UPDAT proceeds to correct the nodal relative power distribution in this manner for each assembly in the core. Upon completion, a final normalization of this relative power distribution is performed to assure its proper normalization to unity.

4.2 Thermal Limit Parameter Evaluation

Having corrected the three-dimensional power distribution of the core based on the measured LPRM readings UPDAT evaluates the thermal limit parameters of the core. The thermal limit parameters based on the corrected power distribution are the average planar linear heat generation rate and the linear heat generation rate, APLHGR and LHGR, respectively.

For each node in the core the average power density of the fuel is calculated as:

$$q_i = \frac{(1. - \gamma) \cdot Q_t}{NPNTS} \cdot P_i^{UPDAT} \qquad (4.2\text{-}1)$$

where γ is the fraction of nodal power produced outside of the fuel rods; Q_t is the thermal power level of the core; and NPNTS is the total number of nodes in the active core region.

With this average power density for each node, the average planar linear heat generation rate is computed as:

$$APLHGR_i = q_i \cdot F_{ax} / L \qquad (4.2\text{-}2)$$

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where F_{ax} is the axial power peaking factor for the node and L is the total length of fuel rods in the node.

The linear heat generation rate is computed as:

$$LHGR_i = APLHGR_i \cdot FINT_i \qquad (4.2\text{-}3)$$

where $FINT_i$ is the lattice power peaking factor for the highest fuel rod in the fuel lattice configuration of node i.

The UPDAT module then compares APLHGR and LHGR values in the node to the APLHGR and LHGR limit for the node and identifies the nodes having the most limiting values.

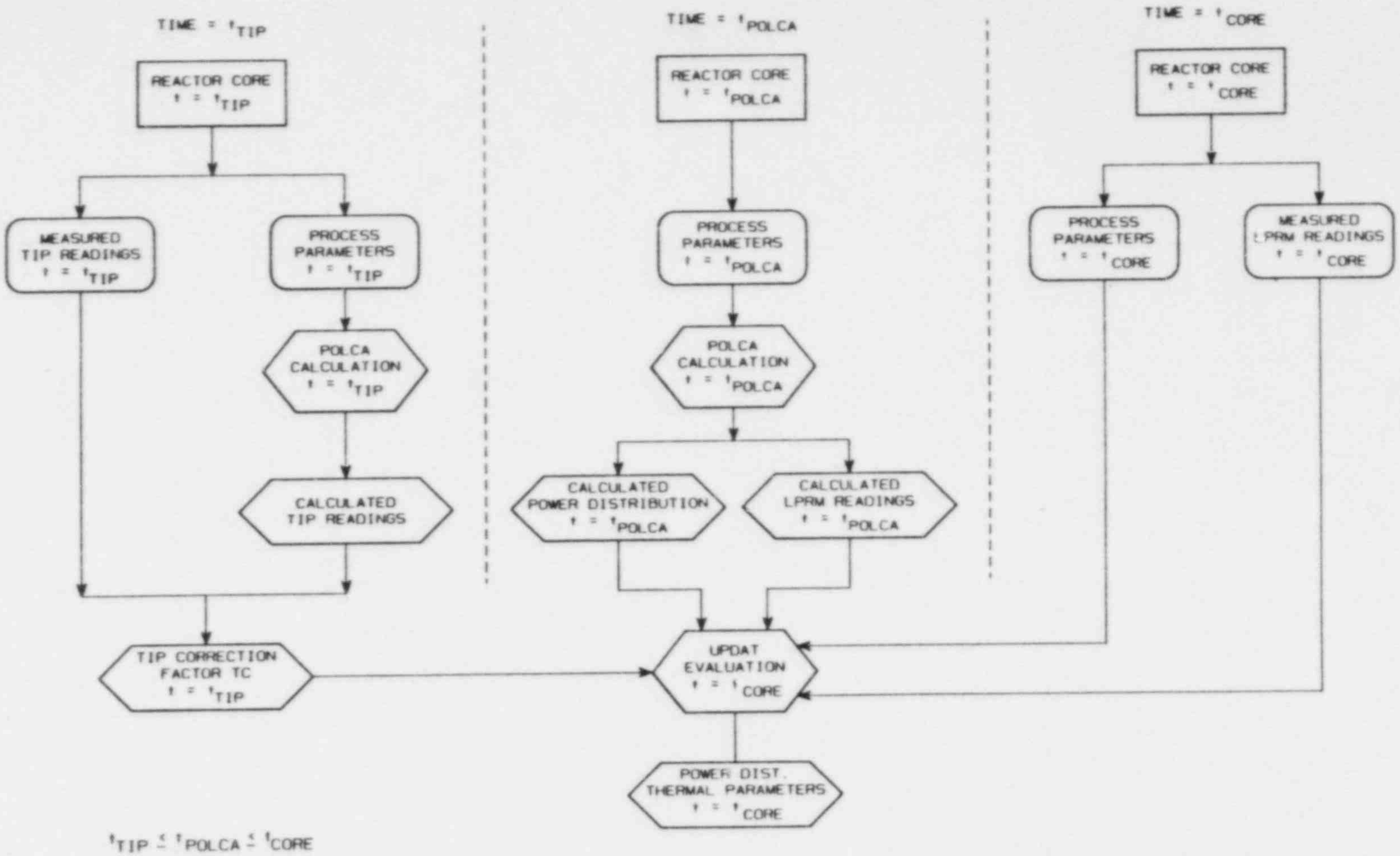


Figure 4-1 COMPUS Operational Sequence

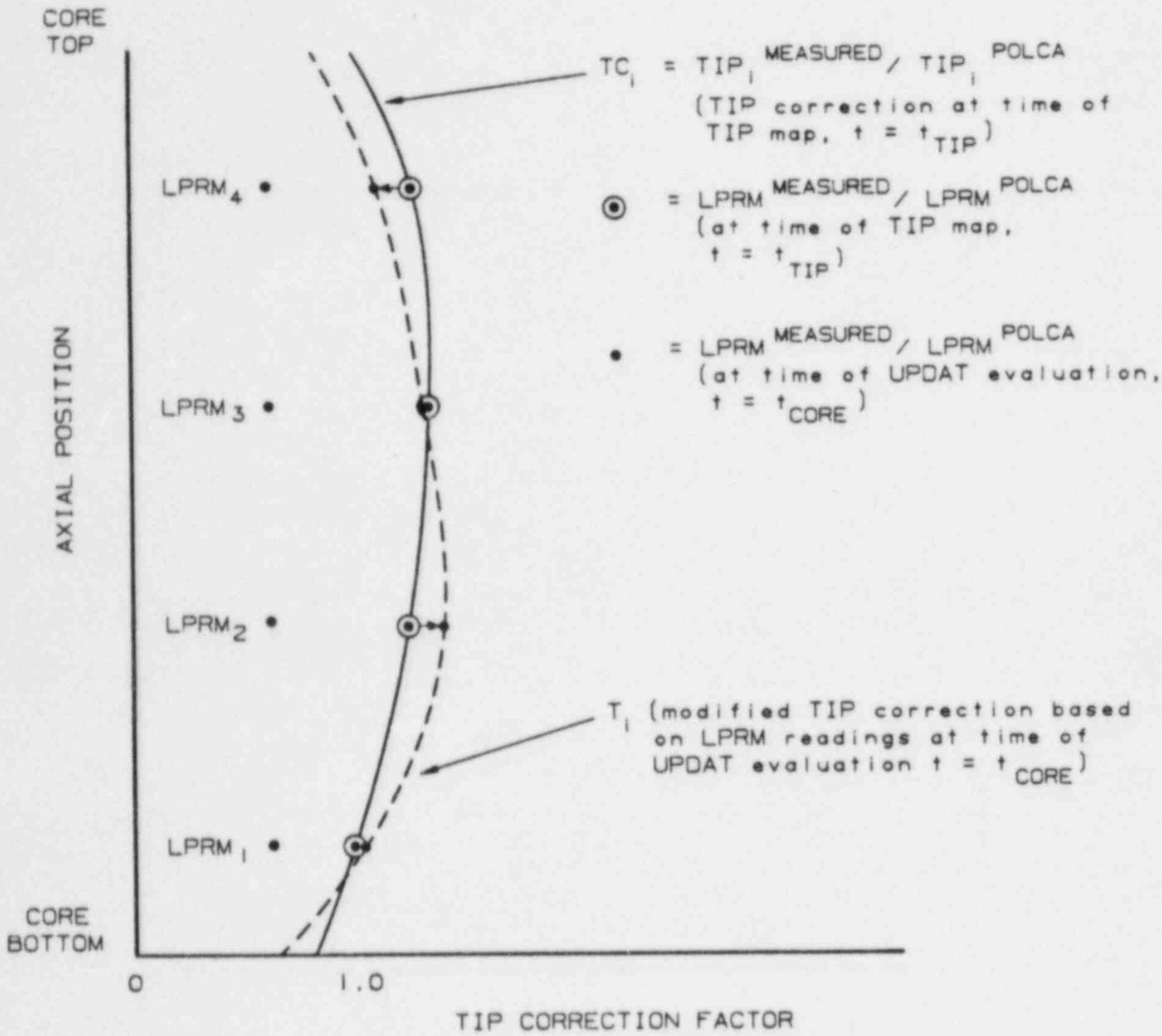


Figure 4-2 Axial TIP Correction Factor

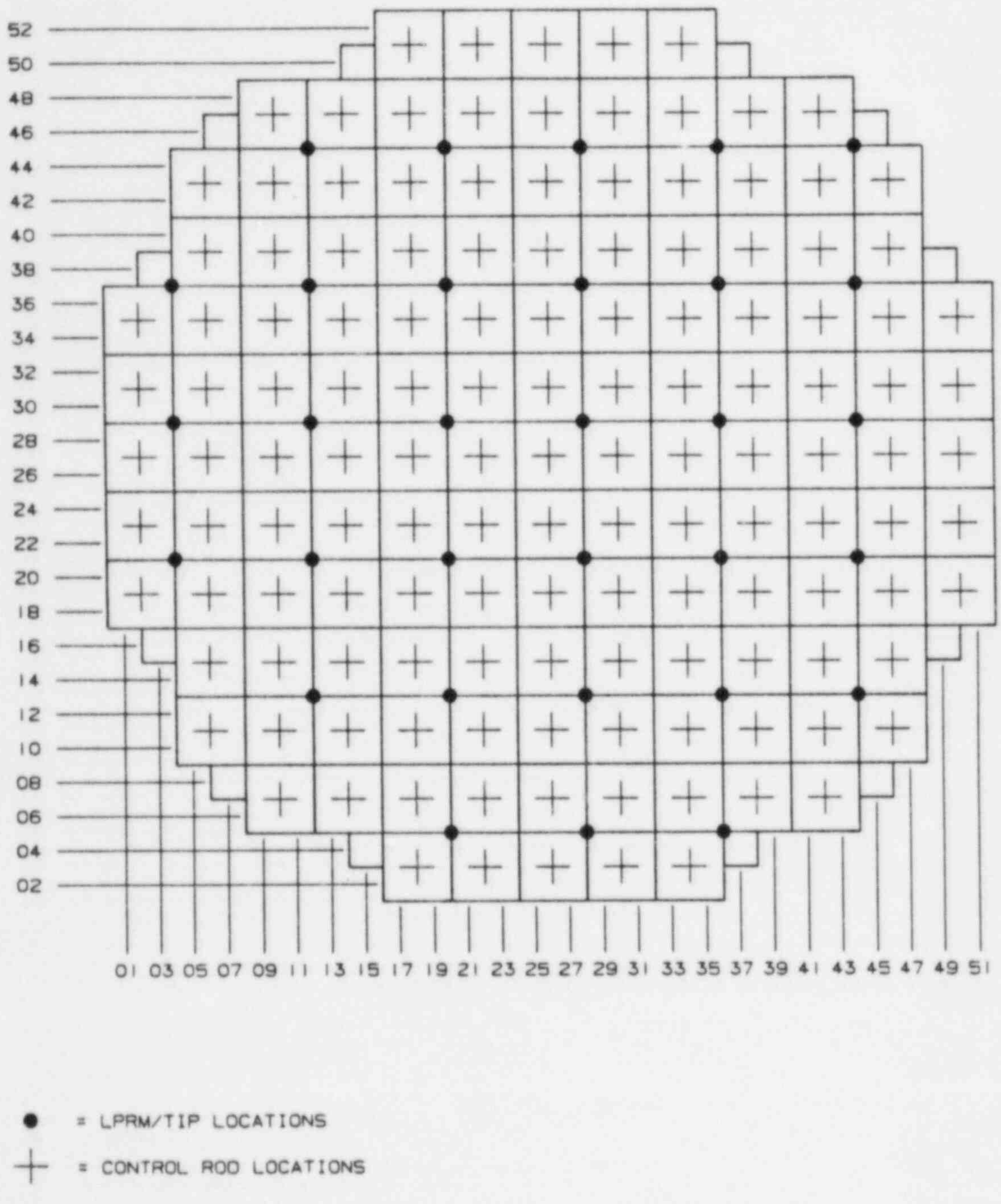
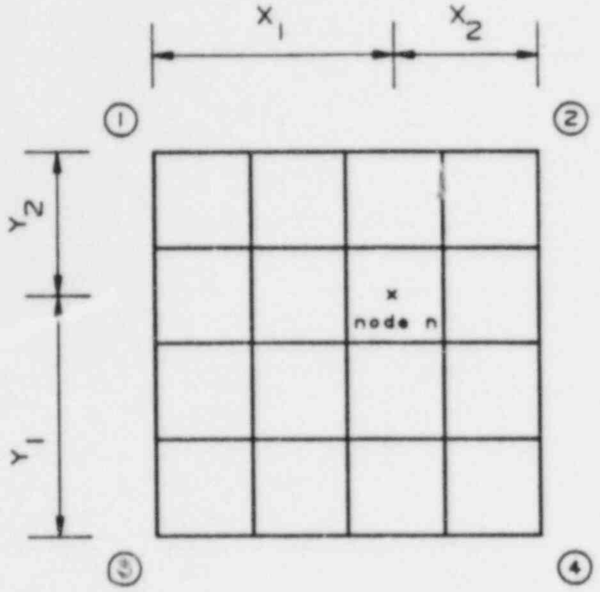
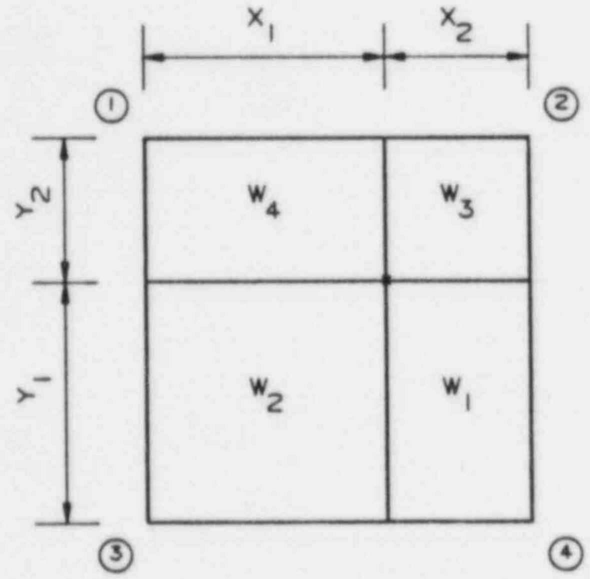


Figure 4-3 Radial LPRM and TIP Locations





RADIAL WEIGHTING FACTORS FOR CORRECTING THE POWER OF NODE N:

$$W_1 = X_2 \cdot Y_1 \qquad W_3 = X_2 \cdot Y_2$$
$$W_2 = X_1 \cdot Y_1 \qquad W_4 = X_1 \cdot Y_2$$

Figure 4-4 Radial Weighting Factors

5.0 UNCERTAINTY EVALUATION OF COMPUS

The primary goal of the COMPUS system is to provide the plant operator with an accurate assessment of the status of the core with regard to margins to thermal limits. Since this information is used to make operational decisions, it is important that the uncertainties which may be present in the COMPUS thermal limit parameters calculations be quantified. The phrase "thermal limit parameters" refers to linear heat generation rate (LHGR), average planar linear heat generation rate (APLHGR) and critical power ratio (CPR). Westinghouse is currently performing a CHF test which will establish the CPR correlation for the Westinghouse QUAD+ assembly. The uncertainty in the COMPUS determination of CPR will be addressed in a future topical when the CPR correlation is available. This section will address the uncertainties in the thermal limit parameters which are dependent on power distribution, i.e., LHGR and APLHGR.

The uncertainty in the COMPUS thermal limit parameters determination has its source in both measurement and calculational uncertainties. Evaluation of thermal limit parameters is performed in the UPDAT module of COMPUS. Examination of the UPDAT equations described in Section 4.0 shows that the sources of measurement uncertainty are the determination of core thermal power level and the in-core nuclear instrumentation readings (TIP and LPRM). The calculational uncertainty in the UPDAT evaluation is made up of the uncertainty in the POLCA power distribution calculation, the uncertainty in the POLCA calculated LPRM and TIP values, and the uncertainty in the UPDAT calculational scheme. The uncertainty analysis described in this section shows that the uncertainty due to the UPDAT calculational scheme is small compared to the overall uncertainties from other sources for cases where use of UPDAT is reasonable.

5.1 COMPUS Uncertainties

COMPUS consists of two main calculational modules: POLCA and UPDAT. As described in Section 3.0, POLCA is a 3-D nodal simulator code which calculates the core power distribution and detector outputs using an explicit core

model. The UPDAT module determines the current power distribution and thermal limit parameters from a previous POLCA power distribution and current measured LPRM and TIP values. The UPDAT calculational process was described in Section 4.0.

The uncertainty in the POLCA power distribution and detector response calculations was determined from extensive comparisons to plant data and gamma scan results and was reported in Reference 2. These uncertainties were used directly in the evaluation of the UPDAT uncertainty.

The core thermal limit parameters are calculated by COMPUS in the UPDAT module. Thus, the evaluation of the COMPUS uncertainties for these calculations involves the determination of the uncertainties in the UPDAT algorithm due to the both calculational inputs from the POLCA module (i.e., power distribution and detector readings) and from plant process measurements including nuclear instrumentation, as well as the error inherent in the UPDAT calculational algorithm itself. The purpose of this section is, therefore, to determine the uncertainties in UPDAT with regard to power distribution and thermal limit parameter calculations.

5.2 Method of Determination of Uncertainties

The determination of the uncertainties in COMPUS was performed by simulating perturbations in core process parameters and evaluating the accuracy of the COMPUS calculation (in the UPDAT module) of the core power distributions and the thermal limit parameters. Uncertainties in calculational and measured inputs to COMPUS (UPDAT) were included through Monte Carlo techniques as described below.

A reference core model was developed using the POLCA module of COMPUS. This model was used to calculate core power distribution, thermal limit parameters, and TIP and LPRM output values at selected core parameters which therefore become the reference case core parameters. Once the reference case calculation had been completed, a second calculation was performed using the

same POLCA model as the reference case but with some changes in the core process parameters. In the case evaluated in this section, flow and power were changed. This case is referred to as the perturbed case.

The uncertainty evaluation involved inputting the core process variables and the calculated LPRM values for the reference case, as though they were measured values, into UPDAT to adjust the power distribution of the perturbed case back to the reference case. Differences between the actual reference case power distribution and the UPDAT determined power distribution thus represent the error inherent in the UPDAT algorithm for the perturbation.

To assess the overall COMPUS (UPDAT) uncertainty, the uncertainties in the inputs to the UPDAT algorithm must be included as well as the calculational uncertainty of the algorithm itself. This was done by characterizing each input variable by a standard deviation and a variable mean. The bases of these standard deviations are discussed in Section 5.3. Monte Carlo trials were run in which the specific value for each input variable was established using a normal distribution characterized by the mean and standard deviation of that variable. The UPDAT calculation was performed with the values of input variables selected in each trial to determine the power distribution and thermal limit parameters corresponding to each Monte Carlo trial. This process was repeated for a sufficiently large number of trials (sets of input variables) for each of the ten highest power nodes such that the overall COMPUS (UPDAT) uncertainty standard deviations for power distribution and thermal limit parameters for peak power nodes were determined with confidence. Numerical results of these analyses are provided in Section 5.5.

5.3 UPDAT Input Uncertainties

The UPDAT algorithm includes both measured and calculated input variables. The measured input variables include the core thermal power and the nuclear instrumentation measurements. Uncertainties from both these sources were obtained from Reference 3. The relative standard deviation used for the uncertainty in core thermal power was 1.76%. The relative standard deviation in the TIP reading was 2.6% while that used for the LPRM measurements was 3.4%.

The calculated input variables in UPDAT are the nodal power distribution values and the calculated TIP and LPRM values. All these values are obtained from POLCA calculations. Uncertainties for the POLCA calculation of these parameters were obtained from Reference 2. The POLCA relative standard deviation in nodal power for average power nodes is [ ]+ This value is also the uncertainty in the LPRM and TIP calculations for average power nodes. For peak power nodes the relative standard deviation in nodal power used was [ ]+ based on gamma scan benchmarking as reported in Reference 2. The uncertainty in TIP and LPRM calculated values at peak nodal locations was [ ]+ (a,c)

An uncertainty in local (rod wise) power distribution was also included in the Monte Carlo analyses. The relative standard deviation for this parameter was [ ]+ based on benchmark calculations reported in Reference 2. (a,c)

A summary of the uncertainties used for the UPDAT input parameters is found in Table 5-1.

5.4 COMPUS Uncertainty Results

The uncertainty in the COMPUS determination of power distribution and thermal limit parameters was evaluated using Monte Carlo techniques. A reference case was run using the POLCA module of COMPUS. The reactor model for the reference case was identical to Quad Cities Unit 1 Cycle 1. The reference case was evaluated at 100% power and 100% flow. A perturbed case was also run in the POLCA module again using the Quad Cities model but with 85% power and 85% flow, all other variables remaining the same as the reference case.

The UPDAT module of COMPUS was then used to "correct" the perturbed case power distribution to the reference case conditions, i.e., 100% power and 100% flow. This was accomplished by inputting the calculated LPRM values from the reference POLCA case as "measured" values in the UPDAT module along with the power distribution from the perturbed case. Uncertainties in LPRM and TIP measurements, LPRM and TIP calculated values, core power distribution and the POLCA calculated power distribution were included through Monte Carlo techniques as described in Section 5.3. Comparisons were made between the

UPDAT "corrected", power distribution and thermal limits and those of the reference case for the ten highest powered node points with 10000 Monte Carlo histories for each of the 10 points. The standard deviations obtained for each of the output variables were averaged over the 10 nodes. Figure 5-1 shows schematically the calculational process for the uncertainty evaluation.

Using the uncertainties associated with average power nodes, the results show an average standard deviation of [ ]+ for the UPDAT nodal power distribution uncertainty, [ ]+ for the APLHGR uncertainty and [ ]+ for the uncertainty in LHGR. Using the uncertainties associated with peak powered nodes, the results show a standard deviation in nodal power distribution of [ ]+ with standard deviations of [ ]+ in APLHGR and [ ]+ in LHGR. The results are summarized in Table 5-2. (a,c)

Additional cases were run to determine the contributions of various parameters to the overall peak node power distribution uncertainty. Setting all the measurement uncertainties to zero (i.e., core power uncertainty, measured TIP and measured LPRM uncertainty) resulted in a standard deviation of [ ]+ a reduction in total uncertainty of [ ]+ Setting the POLCA power distribution uncertainty and the measured uncertainties to zero while keeping the POLCA calculated TIP and LPRM uncertainties at [ ]+ the peak power distribution uncertainty standard deviation is further reduced to [ ]+ The calculational error due to the UPDAT algorithm was obtained by setting all the uncertainties to zero. This error was [ ]+ (a,c)

5.5 Conclusions

The uncertainty in COMPUS power distribution and thermal limit parameters calculations is due chiefly to the uncertainties in the POLCA module in calculating the nodal power distribution and the TIP and LPRM output values. The calculational error in the UPDAT algorithm itself is less than [ ]+ (a,c)

Reference 2 found that the uncertainty in POLCA predicted nodal power was [ ]+ Using the UPDAT algorithm increases this error to about [ ]+ since the uncertainties from measured and calculated LPRM and TIP values must be included. Thus, an uncertainty reduction of about [ ]+ could be obtained (a,c)

through use of POLCA alone rather than both POLCA and UPDAT. This option would neglect corrections from in-core instrumentation. However, procedures could be developed which would use in-core instrumentation to confirm the validity of the POLCA model. This approach is considered a possible future option. Support for this approach is beyond the scope of this topical.

15237.2

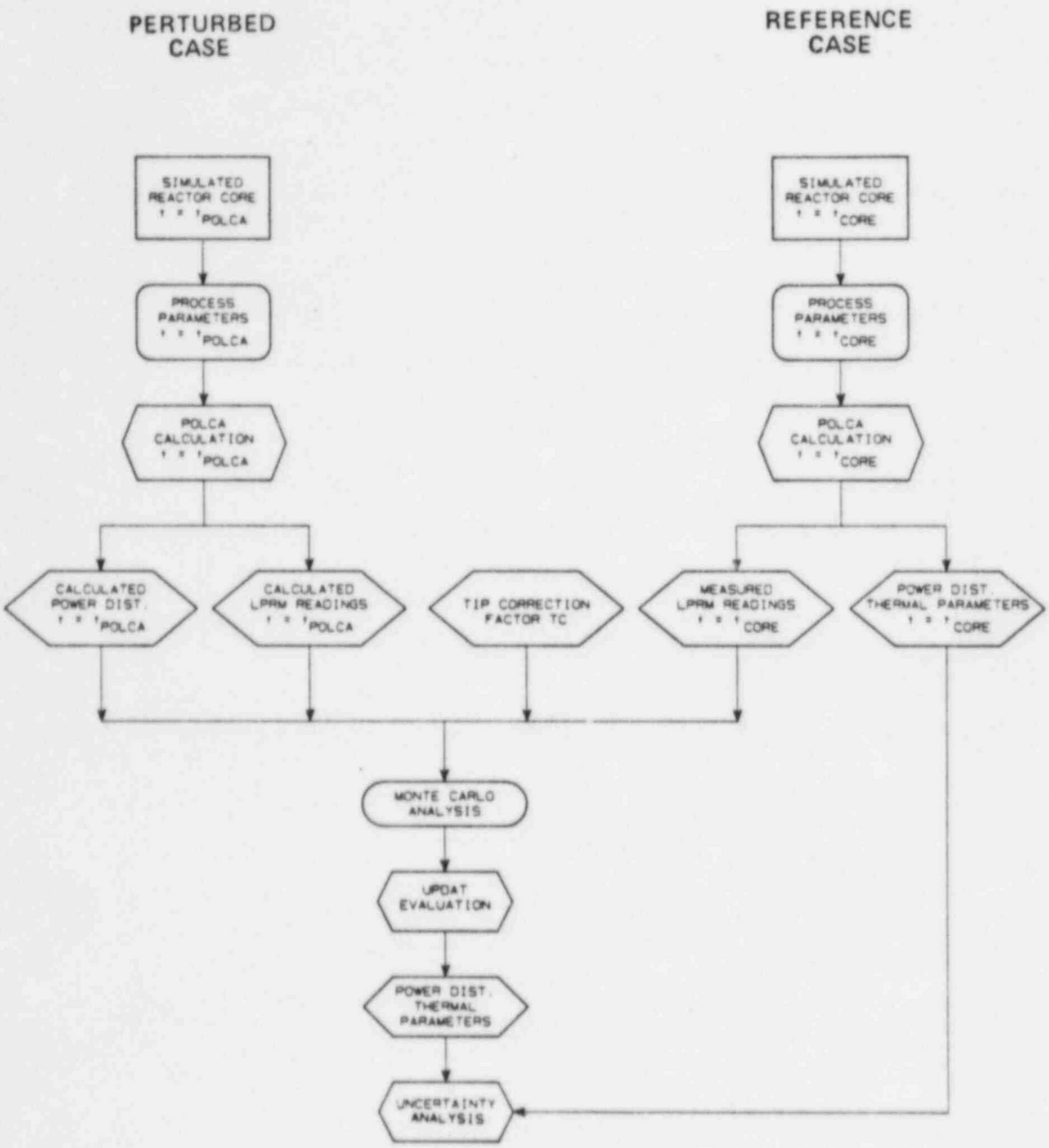


Figure 5-1 Schematic for COMPUS Uncertainty Evaluation

TABLE 5-1
Uncertainties for UPDAT Input Values

Measurement Uncertainties	
Thermal Core Power	1.76%
TIP Measurement	2.6%
LPRM Measurements	3.4%
Calculated Uncertainties	
POLCA Power (Average Node)	
POLCA Power (Peak Node)	
Calculated TIP (Average Node)	
Calculated TIP (Peak Node)	
Calculated LPRM (Average Node)	
Calculated LPRM (Peak Node)	
Local Power	

$[\quad]^{+}$ (a,c)

TABLE 5-2

Uncertainty in UPDAT Calculation

	% Uncertainty (One Std. Dev.)
Power Distribution	
Average Node	
Peak Node	
APLHGR	
Average Node	
Peak Node	
LHGR	
Average Node	
Peak Node	

+ (a,c)

6.0 Conclusion

The COMPUS system provides a state-of-the art on-line monitoring system for boiling water reactors. This system monitors both power distribution and core thermal limits. A 3-D nodal BWR simulator code, POLCA, is used to model local effects. Thermal limit parameters are calculated in the UPDAT module which uses current values of core process parameters including in-core nuclear instrumentation (LPRMs and TIPs) to adjust the last POLCA power distribution to current conditions.

The uncertainty in the power distribution and thermal limit parameter calculations performed by UPDAT was determined through a Monte Carlo uncertainty, analysis for a simulated case involving a 15% change in power and flow. For the peak power nodes of the core the power distribution uncertainty was [ ]+ with uncertainties of [ ]+ and [ ]+ for APLHGR and LHGR (a,c)
respectively. These uncertainties are similar to those of other state-of-the-art core monitoring systems and clearly better than those of earlier systems.

The accuracy of the COMPUS system coupled with its many reactor years of proven use make the COMPUS system attractive for BWR core monitoring.

1980L 6-092485

7.0 References

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