REVIEW OF IN-CORE POWER DISTRIBUTION MEASUREMENTS -- TECHNICAL STATUS AND PROBLEMS

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Prepared by

Nuclear Associations International Corporation
6003 Executive Boulevard
Rockville, Maryland 20852

Principal Investigator
J. R. Fisher

Prepared for

Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, California 94304

Project Manager
A. B. Long

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ABSTRACT

This report reviews the technical status and problems associated with the on-line measurement of core power distributions in nuclear reactors. The need for and significance of the various measured and inferred quantities relating to power distribution are defined. The current status of the instrumentation and data acquisition systems supplied by the four domestic LWR vendors is described. The techniques employed with each of these systems to process the measurements on-line and to infer the different power distribution parameters are also documented.

Sources of uncertainty in the inferred power distribution and problems with the measurement systems are identified and the impact which these may have on plant operations is enumerated. Examples are cited where plant operation has been restricted for these reasons. The report concludes with a summary of areas where additional research and development might be beneficial.

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OVERVIEW AND SUMMARY

This report presents the results of a study conducted by Nuclear Associates International under contract with EPRI to review the technical status and problems associated with nuclear reactor core power distribution measurements. The process and uncertainties associated with the collection and analysis of data to infer the core power distribution are described. Each of the four domestic light water reactor (LWR) vendor's systems are also documented in detail. This basic information will be of interest to reactor engineers and operations personnel.

This prefatory section provides a comprehensive overview to the subject and a summary of existing problems with core power distribution measurements. The need for these measurements is first established and examples are cited of the increasingly restrictive operational requirements which are based upon these measurements. The major components of a power distribution measurement system are reviewed and the significant differences between the existing LWR vendor offerings tabulated. Sources of uncertainty in the inferred power distributions and problems with the measurement systems are then described and examples of resulting plant operating restrictions enumerated. The conclusion to this section summarizes the state of the art of power distribution measurements and suggests areas where additional research and development effort would be beneficial.

INTRODUCTION

The local power production within the core of a commercial light water reactor (LWR) is routinely monitored to satisfy a number of requirements.

These requirements can be categorized by the time frame in which the information is needed; for example:

- Immediate and short term - to assure the integrity of the fuel cladding, the local power production at any axial position in any fuel element must not exceed specified limits. The safety system continuously monitors the global core power to determine that an adequate safety margin exists. Certain conservative assumptions used in the interpretation of this data are routinely verified by in-core detector measurements from which a more detailed power distribution is derived.

- Near term - to plan for and perform reactor control maneuvers in the near term, it is necessary to have local power distribution information in a timely fashion. This data may be used to infer local Xenon distributions, confirm existing reactivity worths, and enable the effect of control actions to be evaluated.

- Long term - to evaluate the performance of the reactor core, its behavior and exposure must be continually tracked. In general, computer codes are used to predict detailed core performance and the measured power distributions are used to confirm and normalize these predictions.

In a modern commercial LWR, the core power distribution is known to be affected by many operating parameters. For example, the shape of the distribution is dependent upon: the core loading; past operating conditions (i.e., localized fuel and poison burnup, fission product creation); recent operating conditions (i.e., localized Xenon buildup); and by present operating conditions (i.e., control rod position, boration level, flow rates). For this reason it is not always adequate to assume a constant power distribution and to then infer the maximum local power density from ex-core detectors or other global power measurements. All of the LWR vendors provide additional in-core instrumentation such as: traversing in-core probes (TIPS), fixed in-core detectors or a combination of both, to "measure" the local power distribution. Depending upon the reactor type, the operating requirements, and the nature of the in-core instrumentation system; the in-core measurements are made on a continuous basis as part of the safety system, on a routine basis in support of operation and administrative controls, or on a periodic basis to ensure that the assumed power distribution is still conservative.

However, it is essential to realize that these in-core detectors do not give a direct measure of local power distribution, but rather the reaction rate (i.e., ^{235}U fission rate or ^{103}Rh capture rate) at the detector. This signal must be corrected for background and detector depletion, and then converted to a value for the power production in the adjacent fuel segments. The conversion relationship is a function of many factors including: past assembly exposure, void or boration level, control rod position, etc. Normally, these factors are generated off-line in the form of polynomials which can be used to make the conversion given the plant operating conditions.

After correction and conversion, the data consists of a discrete number of values for local power at different axial heights (4 or 7 from fixed in-cores, as many as 24 from tip traces) within approximately 20% of the fuel assemblies in the core. If quarter core symmetry exists, it is possible to rotate all this data into the equivalent assemblies in one quarter of the core. In either case, coupling coefficients are used to infer the power in assemblies for which there is no data. The result is the "measured" power distribution within the core.

The intent of this discussion is to indicate that the process of determining the power distribution within a reactor is more complicated than measuring the signals from a number of in-core detectors. In general, the raw data must be processed by an on-line computer at the plant which in turn makes use of parameters from extensive calculations performed on large off-line computers. The details of the entire in-core power distribution measurement process developed by each of the four domestic LWR vendors is the subject of this report.

NEED FOR POWER DISTRIBUTION MEASUREMENTS

As discussed in the preceding section, in-core power distribution measurements are required for a variety of functions including: safety, control and performance evaluation. The importance and extent of these functions have grown in parallel with the increasing size and complexity of nuclear plants and the operating restrictions being imposed upon them. The following examples, which are grouped by reactor type, serve to illustrate some of the needs for in-core power distribution measurements. Additional discussion about these requirements including definitions of measured and derived quantities will be found in Section 1.3, and examples of the actual Technical Specification for measuring and using this information are to be found in Appendix A.

BWR Requirements

Ex-core detectors are not used with a BWR because of: 1) the large separation between the core and the outside of the pressure vessel; and 2) the heterogeneous nature of the core due to coolant boiling. Instead, the prompt response in-core detectors provide input to the safety system as well as the control and core performance monitoring systems. Typical uses of measured data from in-core detectors include:

- The Average Power Range Monitoring system (APRM) derives its inputs from in-core detectors at different axial and radial locations within the core. If the average neutron flux exceeds the limiting safety setting, a reactor scram is initiated. This setting includes a power peaking factor and is established to assure that the Minimum Critical Power Ratio (MCPR) is not violated and therefore that fuel integrity is maintained for all postulated abnormal transient conditions.

- The rod block monitors derive their signals from in-core detectors located in close proximity to the control rod being moved. If the local neutron flux exceeds the trip point, a rod motion inhibit signal is generated.

- The average planar linear heat generation rate, and the local Linear Heat Generation Rate (LHGR) are calculated daily using the in-core instrumentation data. If the specified limits are exceeded, immediate corrective action must be initiated to restore proper operation. These administrative controls will assure that the peak cladding temperature will not exceed specified limits following anticipated operational transients or the postulated design basis loss-of-coolant accident.

- The local power distribution within the core must be closely monitored as part of the Pre-Conditioning Interim Operating Management Recommendations (PCIOMR) in order to minimize the probability of fuel failure. In general, this consists of establishing a LHGR threshold above which the local rate of power increase must be restricted if the specific fuel assembly has not been previously "preconditioned." These recommendations have the effect of restricting BWR load changes if parts of the core are not sufficiently preconditioned.

PWR Requirements

All PWR's are instrumented with some type of incore power distribution measurement system. In general these systems consist of slow time response Rhodium detectors or moveable in-core probes; and are therefore, not suitable for use in the safety system. Instead, ex-core detectors with a fast time response are used as part of the safety system although they do not have the spatial resolution of the in-core system. Consequently, when a global neutron flux anomaly is detected by the ex-core detectors, the in-cores are generally required to analyze the problem in finer detail within a limited time frame, or else, the total reactor power must be reduced by a specified amount. Typical examples of this type of ex-core/in-core power distribution measuring requirements are presented below (examples generally apply to B&W, C-E and W design PWRs):

- The axial offset or flux difference (normalized difference between the upper and lower ex-core detectors in one string) must be within $\pm 5\%$ of the target flux difference for the specific core burnup. If exceeded the offset must be immediately corrected or the reactor reduced below a power level of 90%.

- If the quadrant to average power tilt as measured by the ex-core detector strings exceeds 2%, the peaking factors, F_Q and $F^N_{\Delta H}$, shall be determined within two hours using the in-core detectors or the reactor power shall be reduced by a specified amount.

- The radial and axial peaking factors must not exceed predefined limits, in order to assure 1) that the integrated linear heat flux along the length of a fuel rod does not cause a departure from nucleate boiling in that coolant channel, and 2) that the peak linear power density does not exceed the limit established by the postulated loss-of-coolant accident. These peaking factors, $F^N_{\Delta H}$ and F_Q are calculated from in-core data on a monthly basis, after each core loading change, and when necessitated by ex-core indications.

Impact and Status of Requirements

It is certainly one of the vendor's design objectives to assure that his nuclear steam supply system (NSSS) will operate within the required safety limits and still have the desired operational flexibility and provide the rated power output. However, following the original NSSS design there is an extended period (40-50 years) during which a number of plants incorporating this system will be built, licensed and operated. Experience has shown that during this period the original safety and operating requirements will be modified as the industry's understanding of different technical areas increases and as operating experience is gained.

In terms of the previously discussed requirements for power distribution measurements, more stringent regulations have been imposed which have resulted in: 1) reduced rated power levels; 2) increased times required for plant startup or power increases; and 3) decreased plant load following capabilities. These factors, when coupled with the additional restrictions imposed by the inaccuracies and occasional failures of the power distribution measurement system, have resulted in the reduction of plant availability. Such reductions impose a large financial penalty (for example: a 1% reduction in the availability of a 1000 MWe nuclear power plant is equivalent to a replacement energy cost of ∿$1,000,000/year). Therefore, utilities are interested in the improvement of power distribution measurement systems as one means to counter these problems. The following are recent examples of situations where modified regulations involving core power distributions have led to plant operating restrictions:

- Within the past two years it has been found that flow induced in-core detector assembly vibration may produce unacceptable wear on channels surrounding BWR fuel bundles. The preliminary correction involved the plugging of holes to reduce the coolant flow in the bypass channels in which the detectors are located. However, this produced more voids within the channels which resulted in too much reactivity insertion (void collapse) during a postulated turbine trip without bypass. Consequently, it was necessary to raise the Minimum Critical Power Ratio (MCPR) from 1.19 to 1.4 to ensure fuel integrity during the transient. Subsequent drilling of holes in the lower tie plate to increase coolant flow in the bypass channels permitted the MCPR to be lowered to 1.26 for 7x7 fuel. This power distribution parameter is now 6% more restrictive than it was originally. As an example of the impact of this problem, the Duane Arnold nuclear plant has been restricted to 90% full power since 6/75 when the bypass channels were plugged.

- Recent measurements have shown that fuel rod bowing in W-PWRs can result in an increased enthalpy rise in coolant flowing between bowed rods. To insure that there is not a departure from nucleate boiling it may be necessary to increase the uncertainty associated with the measured $F^N_{\Delta H}$ from 4% to 13%. This will probably impose more restrictive operating conditions on the plant to ensure that $F^N_{\Delta H}$ is maintained within acceptable limits.

- In late '74, Westinghouse performed a generic analysis and showed that most W-PWR's could satisfy the ECCS acceptance criteria limit of $F_Q < 2.32$. As described in Ref. 1, this new restriction could not be satisfied at full power utilizing only the ex-core detectors at the H. B. Robinson plant during cycle 3. Consequently, an automated axial power distribution monitoring system utilizing the movable in-core detectors was used to measure the axial flux profile in two traverse tubes at frequent intervals or when initiated by control rod movement.

When new regulations are promulgated which result in operating limitations, there are a number of interrelated procedures available for correcting the situation and restoring the plant to full operational capability. First, the regulation may include overly conservative assumptions which could be reduced if improved analysis techniques or better experimental data were available. Secondly, a design change may be introduced to alleviate the underlying problem, Third, administrative controls may be established to prevent the situation from arising. Finally, the plant instrumentation, in this case the power distribution monitoring system, may be improved to reduce the measured uncertainty. By monitoring additional variables, increasing sensor accuracy, or modifying analysis procedures, it may be possible to decrease the required margin between the limiting safety system setting and the actual process safety limit. This in turn may increase the operating limits sufficiently to restore the plant's operating capability.

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1. C. W. Crawford, R. H. Chambers, D. B. Waters, "Use of the Axial Power Distribution Monitoring System in H. B. Robinson," TRANSO 19, p. 360 (10/74).

It is this latter alternative which is one of the motivating factors for performing this review study. As will be discussed in more detail in the following portions of this summary section, the different types of power distribution monitoring systems each have certain advantages and certain limitations. It is important that the utilities evaluate the potential impact of these differences on plant operating capabilities, and that suitable research be initiated if it will help to avoid future operating restrictions and possibly improve plant availability.

POWER DISTRIBUTION MEASUREMENT SYSTEMS

An overview of the different types of power distribution measurement systems is presented in this section as a basis for discussion in subsequent sections about: measurement uncertainties, operating problems and possible areas for additional research and development effort. Substantially more detailed presentations about: the instrumentation and data acquisition system are given in Chapter 2; the requirements and uncertainties associated with the measurements in Chapter 3; and the analysis of this information to determine power distributions in Chapter 4.

Each of the major components of the power distribution measurement system is described briefly below. Table I summarizes the specific details of the systems that have been recently supplied by the four different domestic NSSS vendors.

Ex-core Detectors

Generally there are four strings of power range ex-core detectors equally spaced around the periphery of a PWR vessel. Each string consists of two detector segments, the upper one being most sensitive to the top half of the core and the lower one to the bottom half of the core. The detectors themselves are fast response uncompensated ion chambers. Their signals are continuously monitored as part of the safety sytem to detect over power conditions, and axial or azimuthal flux imbalances. The safety system settings include allowances for peaking factors, detector efficiencies and accuracies. These values must be routinely verified by plant personnel from other process measurements including in-core power distribution measurements.

Fixed In-core Detectors

A number of in-core detectors are generally stacked vertically into one assembly and in certain cases a background or alternate detector and a thermocouple for measuring exit coolant temperature are also included. In the GE-BWR the detector assemblies are loaded into approximately one quarter of the narrow/narrow water gaps in the core (a narrow/narrow gap is at the unrodded corners of four fuel assemblies). In the B&W and C-E PWRs the detector assemblies are loaded into or near the center of approximately one quarter of the fuel assemblies. Generally, the locations are selected according to quarter core symmetry so that a certain number of identical locations are monitored in each quadrant. If quarter core symmetry exists, the remaining locations may be rotated into one quadrant to provide pseudo data for most assemblies in that quadrant. Details about specific vendor systems are tabulated in Table I.

Fixed in-core detectors in the GE-BWR are fast response U^{235} fission detectors which provide signals to both the safety system and the power distribution monitoring system supported on the process computer. In the B&W and C-E PWRs, the in-core detectors are primarily slow-response self-powered Rhodium devices. The signals from these devices are provided only to the on-line process computer for analysis and conversion into power distribution measurements.

Movable In-core Detectors

A number of the reactor vendors offer a movable in-core detector (MID) system or traversing in-core probe (TIP). These systems generally consists of multiple independent miniature ^{235}U detectors which can be inserted via indexing mechanisms into a number of separate axial guide tubes. Usually one tube is accessible to all movable detectors in order to permit intercalibration. It is also standard practice to co-locate

TABLE I. COMPARISON OF POWER DISTRIBUTION MEASUREMENT SYSTEMS

REACTOR:	Vendor	B&W - PWR	C.E. - PWR	W - PWR	G.E. - BWR
	Model	177 Bundles, 963 MWe	217 Bundles, 800 MWe	4 Loop 1130 MWe	BWR-4, 1128 MWe
	References	Rancho Seco	St. Lucie	Trojan	Peach Bottom 3
EX-CORES:	# of strings	4	4	4	none
	# of detector/string	2	2	2	-
	detector type	uncompensated ion chamber	uncompensated ion chamber	uncompensated ion chamber	- -
FIXED IN-CORES:	# of strings	52	45	none	43
	% of possible core locations	25%	20%	-	25%
	# of detector/string	7+background	4+background	-	4
	type of detector	Rhodium	Rhodium and Vanadium	-	^{235}U
	length of detector	12 cm	40 cm	-	2.5 cm
	location of string	middle of assembly	middle of assembly	-	corner of assembly
MOVEABLE IN-CORES:	# of traversing tubes	None	19	58	43
	# of independent traversing detectors	-	1	6	5
	type of detector	-	^{235}U	^{235}U	^{235}U
	common calibration tube	-	N/A	Yes	Yes
	fixed in-core calibration	-	19 of 45	none	43 of 43
ON-LINE COMPUTER:	data logging	Yes	Yes	Yes	Yes
	power distribution inferred	Yes	Yes	No	Yes

each movable detector guide tube with a separate fixed in-core detector assembly if there are any in the reactor. This provides the capability for: 1) calibration of fixed in-cores; 2) measurement of the detailed axial flux distribution to supplement the discrete in-core readings; and, 3) substitution of a movable in-core detector for a failed fixed in-core detector.

The signals from the movable in-core detector system are provided primarily to the process computer for logging and analysis.

Calculation of Power Distributions

An integral portion of the power distribution measurement process is the analysis required to correct the raw signals and convert them into power distribution information. As mentioned in the introduction and as described in detail in Chapter 4, this is a complex process involving many localized plant parameters such as fuel burnup, coolant characteristics, control rod locations, etc. The different vendors have chosen to allocate different amounts of this process to the on-line computer. The more elementary on-line systems serve only as a data accumulation device. The information must then be transferred to a larger off-site computer for analysis. The more sophisticated on-line systems are capable of both recording and analyzing the raw data. The calculated power distribution information is normalized to the measured total core power production, and in the case of the BWR is used to iterate on the assumed void distribution. In any on-line or off-line analysis system it is also necessary to supply coupling coefficients and parametric relationships which must be developed in larger design codes such as PDQ.

UNCERTAINTIES AND LIMITATIONS

The requirements for power distribution measurements have been summarized in a previous section for both BWRs and PWRs in terms of the time frame in which the information is used. The potential impact which these measurements may have on plant operation is easiest to see in the immediate or short-term applications. For example, power distribution measurements are used in every LWR to assure fuel integrity either in a closed-loop as part of the safety system or in an open-loop through administrative controls and operator intervention. Clearly, the closer the measured or derived parameters are to the safety or operating system settings during normal modes of plant operation, the greater the probability that measurement system uncertainties or limitations will result in plant operating restrictions. This is a definite problem in existing power distribution measurement systems and consequently there is a need to: 1) reduce the uncertainties in measured and derived variables; and 2) improve measurement techniques in order to eliminate restrictive assumptions.

This section reviews the problems, limitations and uncertainties which arise in the process of inferring the incore power distribution. The major items which will be separately addressed are listed in Table II. They fall into two categories:

- Measurement uncertainties and problems
- Analytical interpretation uncertainties and problems

Where possible, examples are included in the following discussions to indicate the magnitude of the uncertainty or the consequence of the problem in terms of LWR operation. Some of this information was extracted from the responses of results engineers at 17 operating domestic nuclear power plants to a questionnaire dealing with power distribution measurements (please see Appendix B). However, it should be understood that these examples are included for illustrative purposes only. It was not within the scope of this review project to establish quantitatively the magnitude of the various uncertainties associated with the different power distribution measurement systems.

Detector Failure

The failure of detectors which are part of the safety system (for example: PWR ex-cores or certain BWR in-cores) will reduce the number of redundant

TABLE II. UNCERTAINTIES IN THE INFERRED CORE POWER DISTRIBUTION

UNCERTAINTY	IMPORTANCE
A. DETECTOR FAILURES	
• Hard failures - detector lost	• Presently at nuisance level
• Soft failures - erroneous signal	• Sometimes difficult to identify, often times restricts plant operation
B. IN-CORE DETECTOR POSITION	
• Variation in position	• May introduce significant error in BWR
• Asymmetric location in fuel bundle	• Must be corrected for in PWRs with 14x14 assemblies
C. IN-CORE DETECTOR ACCURACY	
• Reproducibility	• Acceptable for individual detectors
• Burnup	• 1-3% per month of full power operation
• Background	• Source of uncertainty in ^{103}Rh detectors
• Drift	• Potential problem dependent upon detector type
D. EX-CORE DETECTOR EFFICIENCY	
• Spatial sensitivity	• Sees only adjacent fuel; hard to detect radial distribution changes; cause of uncertainty in control rod worth measurements
E. DETECTOR INTERPRETATION	
• Rhodium resonance	• Interpretation sensitive to neutron energy spectrum
• Length and number	• Affects prediction of axial flux shape
• Accuracy of conversion and coupling parameters	• Requires extensive off-line calculations to generate.
F. POWER DISTRIBUTION DETERMINATION	
• Core symmetry	• Difficult to maintain
• Interpolation between calculated parameters	• Introduces error for actual operating conditions
• Accurate tracking of exposure	• Possible source of cumulative error
• Approximations in on-line codes	• 2-5% error compared to off-line simulations
• Availability of computer	• Generally good, however many plants have backup capability.

measurement channels and cause an immediate scram or the need to quickly replace the detector. The failure of nonsafety system detectors will decrease the quality of the overall power distribution information and in certain situations cause administrative control problems. In addition, there are Technical Specification requirements for a minimum amount of nonsafety in-core instrumentation in order for the plant to operate. For example:

- Above 80% power in a B&W PWR there must be a specified number of operational detectors at different heights and in different quadrants of the score (see Section 3.3.2 and Reg. 2).
- At least one C-E PWR has the requirements that 75% of the in-core detector strings have at least three of the four detectors operational (see 3.4.2 and Ref. 3).
- Above 90% power in a 4-loop W-PWR, there is a requirement that a certain number of traversing thimbles and the associated instrumentation be operational (see Section 3.2.2 and Ref. 4).

Approximate detector failure rate data were calculated from the questionnaire responses and are summarized and compared to reported European information in Table III.

TABLE III. DETECTOR FAILURE RATE DATA

Detector Type	# of Reactors	# of Detectors	% Failure Per Year	Other Data (Ref. 5)
<u>BWRs</u>				
Fixed In-cores	5	572	5%	7%
Moveable In-cores	5	19	37%	-
<u>PWRs</u>				
Ex-cores	8	64	5%	-
Fixed In-cores	4	724	2%	2.5%
Moveable In-cores	4	17	70%	-

The results suggest that the failure rates of ex-core and fixed in-core detectors do not cause an excessive problem (excluded are the few instances

2. Rancho Seco Final Safety Analysis Report, Sacramento Municipal Power District (Revision 2/75).
3. St. Lucie Final Safety Analysis Report, Florida Power and Light (Revision 8/75).
4. Trojan Final Safety Analysis Report, Portland General Electric.
5. H. Bock, "Miniature Detectors for Reactor In-core Neutron Flux Monitoring", Atomaic Energy Review, 14, 1, p. 87, (1976).

where large numbers of fixed in-core detector failures resulted from design or operating errors). The much shorter lifetime of moveable in-core detectors is due to greater operational stresses, but because replacement is relatively easy and can be done at power, the higher failure rate seems only to be at the nuisance level. These conclusions are supported by the majority of the questionnaire respondents who felt that although detector reliability was not as good as was promised it was still acceptable.

A more serious problem seems to be caused by "soft" detector failures, where it is not immediately obvious that the detector output is bad. There have been instances, for example, where a faulty moveable detector was discovered by off-site personnel when measured and predicted results were compared. The site personnel were then able to substantiate the error by traversing the suspect and another detector in a common tube and comparing the measurements. The process of demonstrating that a suspect signal is the result of a soft detector failure, and not an anomalous in-core condition, is not always easy. Inter-detector calibration capability will obviously facilitate the task of fault isolation. However, the process is still time-consuming and the plant may be restricted until the detector output can be shown to be erroneous.

Detector Location

Small variations in detector location may produce large changes in the measured reaction rate if there are significant spatial gradients in the radiation being monitored. This appears to be a particular problem for BWRs because of the variation in the thermal neutron flux across the water gap in which the detector assembly is located. Small detector displacements or variations in the actual gap width appear to cause a big difference in the measured ^{235}U fission rate. Reference 6 attributes a 2-5% uncertainty in the measured BWR power distribution due to variations in detector position. However, recent measurements at a number of BWRs in which TIP readings from supposedly symmetric locations were compared suggest that the integrated readings (over the axial height of the core)

6. J. F. Carew, "Process Computer Performance Evaluation Accuracy," NEDO-20340 (6/74).

may differ by 15% and that point readings may differ by as much as 30%. The magnitude of this problem is typified by apparently anomalous tip readings which are currently limiting the Hatch Nuclear Plant to 94% of rated power.

In contrast, PWR power distribution measurements are not subject to large detector position related uncertainties. This is because PWR in-core detectors are better constrained within the center of fuel assemblies away from large water gaps. However, the absolute location of the detector string within the fuel assemblies must be properly accounted for in subsequent data analysis. For example, in a 14x14 fuel assembly the detector string is located off center and it is necessary to correct for the assembly orientation relative to the flux gradient when comparing measurements from otherwise symmetrical locations.

Detector Accuracy

In-core instrumentation consists primarily of ^{235}U fission detectors and self-powered ^{103}Rh neutron capture detectors. In principle both are capable of measuring relative neutron flux levels to better than one percent. The ^{235}U detector is relatively insensitive to background because of the large signal produced by fission events. In self-powered detectors the signal to noise ratio is much poorer and a gamma background detector is generally included in every in-core detector string. As the ^{103}Rh in the detector burns up, the background signal may increase to 30% of the neutron signal. This has been cited by a number of utilities as a source of erroneous azimuthal tilt indications.

The burn-up rate for self-powered neutron detectors is approximately 1.2% per full power month of operation and it is about three times faster for ^{235}U detectors.[5] The resulting change in detector efficiency can be obtained by effectively integrating the detector signal over time in order to account for the depletion effects. If inter-calibration capability exists, then the burn-up correction can be periodically

5. H. Bock, "Miniature Detectors for Reactor In-core Neutron Flux Monitoring," Atomic Energy Review, 14, 1, p. 87 (1976).

checked. If inter-calibration capability does not exist it is necessary to: 1) establish the absolute sensitivity of the detector prior to installation; 2) devise an accurate technique for determining the change in sensitivity as a function of neutron exposure.

Although detector accuracy should not be a problem in principle, it does cause problems in practice. For example, in BWRs it has been observed that fixed in-core detector signals may drift with time. This is felt to be a result of insulation leaks which allows the composition of the detector fill gas to change with time. Significant changes in signal have also been observed during the long time period ($\sim$ one hour) required to run a full core map with a moveable in-core detector. This may reflect a nonsteady-state condition in the core and/or drifts in the actual detector signal. Some utilities have found it necessary to renormalize the detector readings partway through the flux map, and Westinghouse is validating a quarter core mapping procedure[7] in order to reduce the measurement time and thus the uncertainty.

Ex-Core Detector Efficiency

The ex-core detectors in a PWR serve to monitor the neutron power level as part of the safety system and to detect axial and azimuthal flux balances. These detectors are primarily sensitive to the power distribution in the most adjacent outer fuel assemblies. A recent analysis[8] determined that 93% of the signal was derived from fissions in the seven nearest fuel elements. Consequently, it is necessary to adjust parameters derived from ex-core data to account for factors like radial peaking which are not measured. To ensure that this constant adjustment is conservative for all permitted operating conditions, it is further necessary to increase the margin between the safety system setting and the safety system limit.

7. K. A. Jones, R. A. Kerr, "Quarter-Core Flux Mapping - Power Distribution Measurement Techniques for Non-Steady State Conditions in Westinghouse PWR Cores," TANSAO 23, p. 481 (6/76).
8. M. W. Crump, J. C. Lee, "Calculations of Spatial Weighting Function for Ex-Core Detectors," TANSAO 23, p. 461 (6/76).

For example, margins of 15% have been included to assure that the value of F_Q derived from ex-core detectors is conservative. Recent evaluations of a three level ex-core detector system by C-E has permitted this margin to be reduced to 7.5% which is still larger than the 5% uncertainty associated with values determined from in-core data.[9]

A related problem exists when ex-core detectors are used as part of a reactivity computer to measure the worth of control rods. Depending upon the relative position of the detectors and the control rods, variations of up to 25% have been observed between the computed reactivity worth and the value obtained from boron end point measurements. This difference is attributable to a change in effective ex-core detector efficiency during the rod withdrawal due to the spatial redistribution of the flux in the core.

Detector Interpretation

Before the core power distribution can be determined and the plant operating conditions compared with operating limits, it is necessary to convert the measured ^{235}U (n, f) or ^{103}Rh (n, β) reaction rate in the detector to the local heat generation rate in the adjacent fuel elements. This conversion is a function of many factors including the past exposure history of the fuel segments, moderator conditions, control rod insertion, etc. Since the integral of the inferred power distribution over the entire core may be normalized with the independently measured total core power production, the uncertainties of interest are second order and arise from the inability of the conversion process to properly account for spatial or energy dependent effects.

Comparison of inferred power distributions with the actual distribution as determined by gamma scanning of the irradiated fuel assemblies, would yield a direct indication of total uncertainty. Unfortunately, adequate data of this type is not currently available in the open literature. What have been published are: 1) comparisons between the off-line simulated power distribution and the on-line inferred distribution using

9. J. R. Humphries, A. C. Kadak, R. W. Kanpp, "On-Line Power Distribution Monitoring with In-core Detectors," TANSAO 22, p. 647 (11/75).

a pseudo detector signal generated from the off-line simulation[6, 9]; 2) comparisons between off-line simulated power distributions and the on-line inferred distributions[10, 11]; and 3) comparisons between off-line simulated power distributions and gamma scans of the indicated fuel assemblies.[11] Typical values for these quoted uncertainties are shown in Table IV, and from them the following generalizations may be drawn:

- The approximations used in the on-line analysis codes introduce a small uncertainty (∿3%) when compared to the more detailed off-line calculations.
- The power distribution parameters predicted by the off-line simulation codes are quoted to be within 5 or 6% of the values inferred from in-core measurements.
- The power distribution parameters predicted by the off-line simulation codes are within 10% of the values measured by gamma scan techniques.
- Technical specifications require that certain power distribution parameters derived from in-core data be increased by 5-10% to account for measurement uncertainties.

Until better comparison exist between power distributions inferred from on-line data and power distributions measured by gamma scan techniques, caution must be exercised in extrapolating simulator uncertainties to the on-line situation. During actual plant operation, decisions must be made on the basis of results provided at that time. In particular, erroneous detector information is often hard to identify and even more difficult to discard. Consequently, additional errors and uncertainties may have to be tolerated at the time, even though subsequent analysis can correct for the problem.

On-Line Power Distribution Calculations

The on-line computer calculates the power distribution by first inferring the nodal fuel assembly power from adjacent in-core detectors and then

6. J. F. Carew, "Process Computer Performance Evaluation Accuracy," NEDO-20340 (6/74).
9. J. R. Humphries, A. C. Kadak, R. W. Knapp, "On-Line Power Distribution Monitoring with In-core Detectors, TANSAO 22, p. 647 (11/75).
10. T. G. Ober, P. H. Gavin, "Operational Experience with Self-Powered In-core Nuclear Instrumentation," TANSAO 19, p. 218 (10/74).
11. G. R. Parkos, "BWR Simulator Methods Verification," NEDO-20946 (5/76).

TABLE IV. QUOTED UNCERTAINTIES FOR POWER DISTRIBUTION CALCULATIONS

EXAMPLE	REFERENCE	DIFFERENCE AVE.	DIFFERENCE STD.
Case 1: Simulator vs. On-line Analysis Using Simulator Generated Detector Signals			
• Nodal Power	GE Ref. 6	-2.6%	3.5%
• F_Q calculator	C-E Ref. 9	-1.0%	2.6%
Case 2: Simulator vs. On-line Analysis			
• Axial peaking factor	GE Ref. 11	+0.2%	3.4%
• Axial peaking factor	C-E Ref. 10	--	2.0%
• Radial peaking factor	GE Ref. 11	-5.0%	6.0%
• Radial peaking factor	C-E Ref. 10	--	2.5%
Case 3: Simulator vs. Gamma Scans			
• Axial peaking factor, partially controlled core	GE Ref. 11	+9.7%	10.5%
• Axial peaking factor, uncontrolled core	GE Ref. 11	+3.6%	5.7%
• Radial peaking factor	GE Ref. 11	+0.7%	3.2%
Technical Specifications - Measurement Uncertainties			
• F_Q - Westinghouse PWR	Ref. 4		5%
• F_Q - Combustion Engineering PWR	Ref. 3		8%

developing a detailed distribution using coupling coefficient and in general assuming core symmetry. The different vendor systems use slightly different approaches with the BWR calculations being the most complex due to the need to iterate on the coolant void distribution. The accuracy of any of these techniques depends upon how well the coupling coefficients and other parameters, calculated with off-line codes, account for the particular set of conditions under which the plant is operating. The derivation of these coefficients and parameters is not an easy task, as is discussed in Chapter 4. In many cases the utility is dependent upon the vendor for these values.

The following two examples indicate the type of approximations which may introduce uncertainty into the calculated distributions:

- Conversion of a rhodium detector signal to a nodal assembly power requires an assumption about the neutron energy spectrum relative to the ^{103}Rh capture resonance. There is some evidence that the presence or absence of a control rod may shift the spectrum and significantly affect this conversion.

- In the on-line calculational scheme used by GE, the most recent TIP trace in a channel is normalized by the current LPRM data from that channel and used to calculate the power distribution. In many instances, the actual power shape will have changed since the TIP trace was recorded and consequently the calculated distribution will be in error. This situation can be corrected by rerunning the trace when the discrepancy becomes too large.

The role and dependence upon on-line calculational capability has increased in parallel with the need to have more precise information about the core power distribution. Although few Technical Specifications impose a direct restriction on plant operation if the computer is not available, many require an increased level of surveillance and analysis that is very demanding upon the operational staff. Consequently, many utilities, especially BWR owners, have implemented a redundant on-site or back-up off-site computer. The respondents to our questionnaire generally indicated that if the back-up facility were also not available; the plant might be restricted in its mode of operation, prevented from starting up, or derated in power; and that additional operating personnel would certainly be required.

Additional characteristics of the on-line power distribution measurement systems at seven operating BWRs and eight PWRs were rated by the cognizant results engineers in the responses to our questionnaire. This information is tabulated below and from it one can draw a number of general conclusions.

TABLE V. EVALUATION OF ON-LINE SYSTEM
(Only "Good" and "Poor" responses tabulated, the remainder were "Adequate")

	BWR		PWR	
	% Good	% Poor	%Good	% Poor
Detector Reliability	42	28	14	28
Detector Accuracy	14	14	38	28
Composite evaluation of performance calculations	57	7	75	0
Satisfaction with calculations	28	0	50	25
Ability to obtain conversion parameters	40	0	50	12
Time required for on-line calculations	0	43	50	0
Capability to transfer data to off-line system	0	71	12	25
Adequacy of information presentation	28	28	50	0
Availability of on-line calculational system	71	14	50	12

First, the majority of the site engineers are satisfied with: the performance calculations, availability of the on-line computer, and the ease of obtaining conversion parameters. Second, certain aspects of some systems require improvements; for example: detector accuracy and reliablity, calculational time, information presentation. Finally, the majority of the respondents indicated that the process of transferring information to the off-site computer was inadequate.

CONCLUSIONS

Core power distribution information is required to satisfy a number of fundamental reactor safety and operational requirements. For example, all commercial LWRs must operate within prescribed limits based upon this information to assure fuel integrity and thus protect the safety of the public and the capital investment in the plant. Similarly, this information is used to evaluate reactor control actions and track the performance of the fuel. To satisfy these and other safety, control, and surveillance requirements, the core power distribution measurement system must be capable of collecting and processing in-core and ex-core data within the appropriate time frame. As described in a preceding subsection, this includes providing information:

- on a continuous basis to the safety system to assure that safety limits are not exceeded,
- on a routine basis to maintain the plant within operating limits which would protect the fuel should a transient occur,
- on a near-term basis to monitor recent reactor control actions and evaluate future ones,
- on a long-term basis to track fuel performance.

Experience has shown that the number and complexity of the core power distribution requirements have expanded as: reactor operating experience has grown, plant size has increased, and tolerances have been reduced to improve operating efficiency. As an illustration of the types of factors which might affect these requirements, one need only recall recent design operations problems or expanded safety analysis considerations such as: fuel densification, fuel bowing, pellet-clad interactions, internals vibration, loss-of-coolant accidents, anticipated transients without scram. Each of these examples has resulted in interim or permanent changes to the limits imposed upon core power distribution, not only in new reactors, but also in many operating reactors. Although in some cases the requirements have later been reduced or removed by design modifications, the trend has caused a continual decrease in the margin between the normal operating range and the imposed safety or operational limits of the plant.

Consequently, the core power distribution measurement system has become one of the key factors in determining whether these new requirements can be met without the restriction of plant operation. There are numerous examples, some of which have been described previously, in which availability or capacity of the plant has been derated because of limitations or uncertainties in such measurement systems. In the following sections of this report, the various LWR vendor-supplied power distribution measurement systems are described in detail. From these descriptions, and as discussed in the preceding subsection, it is apparent that there are different types of limitations and uncertainties associated with each design. To illustrate this point the following four designrelated restrictions are cited:

- The lack of inter-calibrational capability for in-core detectors (for example, a moveable in-core detector system) makes it difficult to resolve anomalous detector readings.

- The lack of fixed in-core detectors makes it difficult to monitor core power distribution during nonsteady-state conditions. Because ex-core detectors only monitor global power shapes, overly restrictive operating requirements may have to be imposed to assure local limits are not exceeded.

- Slow response Rhodium detectors cannot easily be used as part of the safety system; their signal to background ratio can become a problem with burnup; and the interpretation of their signal is sensitive to changes in neutron energy spectrum.

- The location of a thermal neutron detector in a significant water gap can introduce uncertainties because of spatial flux gradients if the detector position or gap size varies.

The underlying conclusion is that the flexibility and the limitations of the core power distribution measurement system must be carefully reviewed as part of the plant selection process by a utility. Attention must be paid to: 1) the margin between normal operating conditions and safety or operational limits; 2) restrictions which may arise as the plant gets older and its mode of operation changes: 3) the adaptability of the measurement system to new monitoring requirements; 4) the uncertainties and operating problems associated with the measurement system which may in themselves cause operating restrictions.

In addition to documenting the nature and limitations of the systems currently used for measuring core power distribution in LWRs, this study has identified certain generic problem areas which may be suitable for additional research and development effort. The major areas are discussed separately in the following paragraphs. The reader will note that a number of currently significant problems have been omitted because they appear to be primarily design dependent and thus better left to the vendor for correction. In a number of these cases we are aware that corrective actions are being evaluated that will hopefully eliminate the problem.

Quantification of Uncertainty

Operational experience at numerous LWRs suggests that there are larger uncertainties associated with power distribution measurements than have been predicted by the vendors. These predictions have been based largely on comparisons between the measured distributions and off-line simulated power distributions. Unfortunately, no comparisons have yet been reported between the power distribution inferred on-line and an independent measurement of the distribution such as can be derived from gamma scans of the fuel immediately after shutdown. However, it may soon be possible to make such a comparison on the basis of the Quad City data (EPRI Report NP-214) and other gamma scan programs underway.

In order to reduce the margin of uncertainty between the inferred and the actual power distribution it is first necessary to quantify the existing uncertainty. This could be done in a two-part program. First, the measured detector signals (from ex-cores, fixed in-cores and moveable in-cores) and the inferred power distributions should be recorded as part of any future gamma scan power distribution measurement programs. This data would serve as the basis for a more realistic assessment of the accuracy of on-line inferred power distributions for a limited number of core configurations. Coupled with this a second program could be initiated to routinely document various measured and inferred power distribution parameters throughout the fuel cycle at a number of different LWRs. The data from each plant could then be analyzed for consistency and compared to the appropriate simulated core power distributions. From this latter program it would be possible to quantitatively establish specific sources of uncertainty such as: signal to background versus detector location and burnup; reproducibility of individual detector signals, consistency of inter-calibration ratios, variation in measurements from symmetric locations, agreement between inferred and simulated distributions, etc.

Identification of Soft Detector Failures

The soft failure of a detector has been cited as a common problem that often results in unnecessary loss of plant capacity. In such situations, the detector generates an indication of an anomalous flux tilt or peak which must be treated as "reality" until the signal can be proven to be erroneous. Generally, early indications of this problem go undetected, and it is not until an operating or safety limit is reached (i.e., rod block, quadrant tilt, F_Q or $F^N_{\Delta H}$ peak) that an evaluation effort is started. Even the most direct process of inter-calibration with a moveable detector is time-consuming, and in some systems this capability does not exist.

The operating penalties associated with soft failures of detectors appear to justify the development of a system for automatically tracking individual detectors in order to provide an early indication of erroneous behavior. Such a system could record and utilize information about past performance relative to inter-calibration detectors, symmetrical detectors and neighboring detectors and from this estimate the validity of the present signal.

Generation of Factors Used in On-Line Analysis

The factors and coefficients that are used to interpret the detector signal and subsequently infer the core power distribution are generated off-line by complicated analytical procedures similar to those used in core physics analysis. Inherent in the generation and use of these factors are numerous assumptions including: analytical approximations to the geometry and neutron energy representation, judgements about the best means for mocking up the surrounding environment, or selection of the cases from which the parameter are derived. It is evident that these assumptions and judgements will impact the power distribution inferred in the on-line analysis.

Since it is the derived power distribution which governs actual reactor operation from the standpoint of safety to the choice of operating strategies, it is vital that utilities understand these analytical process and assumptions in detail. Therefore, it may be desirable to incorporate the appropriate modifications into the EPRI nuclear code package to facilitate the generation of these factors and coefficients. In addition, sensitivity analyses should be performed to establish the impact of the various assumptions and judgements on the accuracy and uncertainty of the predicted core power distributions. This task should be closely coordinated with the project to quantify the overall certainty of these measurements described previously.

Evaluation of Power Distribution Related Limitations

The growing number and complexity of the limitations placed upon core power distribution continue to increase the industry's dependence upon on-line measurement systems. As the body of this report discusses, quite different philosophies are inherent in the monitoring systems offered by the different LWR vendors. These include: differing emphasis on ex-core vs. in-core instrumentation, differing types and numbers of in-core detectors, and differing on-line analysis capability. In addition, the designs of future measurement systems are being modified to correct for some of the present limitations as well as to provide additional flexibility and measurement capability. All these factors have potential impact upon the ability of the system to meet the operating requirements of the plant.

Because of the fundamental importance of adequate core power distribution monitoring capability, it would be valuable to evaluate the impact of different measurement system designs and their associated uncertainties on the ability of plants to meet existing and future operating restrictions. The results from such a project would be a more definitive appraisal of the type of systems which will enable LWRs to maintain the greatest operating flexibility in the future.

Alexander B. Long, Project Manager
Nuclear Safety and Analysis Department
November 22, 1976

1.0 DEFINITION OF NEED FOR IN-CORE POWER DISTRIBUTION

1.1 Early Experience with In-Core Power Distribution Data

In order to better understand the current need and status for determining the in-core power distribution, it will be helpful to review in summary fashion the evolution of current and near future reactor designs.

The very early reactors were research, test or demonstration plants such as MTR, ETR, VBWR, Shipping-port, Big Rock, Elk River, etc. All these reactors had small cores compared to today's plants and the monitoring and safety systems instrumentation was located external to the core. The inherent assumption was that these external instruments could "see" the average response of the core. That is, that the neutron flux at the detector was directly proportional to the average flux in the core (or core power level) for all operating conditions. Hence, any change in core flux would result in an immediate and corresponding change in the flux at the detectors. In addition, the designs, insofar as power peaking and related quantities were concerned, were very conservative so that high performance of fuel was not often encountered and the actual margins between normal operating conditions and the operating limits imposed to protect against fuel damage and possible consequences relative to design accident considerations were large. In addition, these cores tended to be relatively simple and more amenable to analyses.

Even in these smaller cores it was found that the out-of-core detectors were sensitive to operating conditions and had to be periodically re-calibrated by a heat balance in order to reflect true core power. Sources of the error in indicated power at the instrument locations were such items as: change in moderator density between the core and the instrument; changes in the axial power shape due to control rod configuration changes and/or moderator density changes within the core. These effects were particularly troublesome in the BWR's. In the PWR's the use of soluble poison and the almost constant moderator density did help to minimize the problem. This difficulty in monitoring of average core power was one motivating force towards the incorporation of in-core instruments for BWR's.

In addition, there was a strong interest in obtaining more information about the detailed core behavior in order to verify the analyses which were being done; this was true for both BWR's and PWR's. Some of the early in-core instrument techniques included the use of irradiated (FLUX) wires and traversing fission or ion chambers. The output from these systems did not interface directly into any of the monitoring systems but provided information which was factored into core design and operation on a longer term basis.

The difficulty experienced in monitoring the reactor cores and the interest of design and operating staffs in obtaining more information on the in-core power distribution were two of the factors that affected the core monitoring approach. In addition, power reactors were evolving from the prototype and demonstration stage to the commercial stage where the economics became important. At the same time computers and hence analytical techniques were becoming more sophisticated thus allowing the designer to optimize the designs and reduce the conservatisms.

Since these design considerations impact on the need for information on the in-core power distribution, and eventually on the approach for obtaining this information, it will be worthwhile to digress and review briefly the evolution of current designs and considerations that affected them.

1.2 Design Evolution

The total power that can be obtained from a given core at a given time; i.e., a given number of fuel pins, is governed by what the peak pin or peak pin segment experiences. The operating limits that are set for the fuel of any core are such that no portion of the fuel may exceed these limits. Hence, whenever the peak of the core reaches an operating limit the total core power cannot be raised any further, even though most of the core is well below the operating limits. Because of this there is a strong economic incentive to make the peaks in the core close to the average conditions. Also, economic considerations dictate that it is prudent to reduce the magnitude of the very conservative margins between the operating limits and actual operating conditions. This can be done by improving the monitoring systems such that the uncertainty in actual operating conditions are reduced. These two incentives:

- Reducing the Peak to average in the core
- Removing some of the uncertainty in the actual operating peak have

had a significant effect on the core design and core monitoring systems as they evolved to current state-of-the-art.

There are many reasons for these neutron flux peaks (and hence power peaks) in a reactor core. A few of the more common causes are:

a) Control Rods - cause a flux depression and consequently a peak away from the rod.

b) Water Gaps - neutrons are thermalized in water gaps resulting in a high density of thermal reactions near a gap; and a higher rate of fission for fuel pins near a gap.

c) Leakage - neutrons that leak from the core surface are lost and the neutron density near the core periphery is reduced, and hence the power. To compensate the neutron density somewhere else in the core increases.

d) High Reactivity Regions - since there is more fission in high reactivity regions, the neutron density tends to be higher near these regions.

e) Non-uniform Moderation - neutron density tends to be highest when the moderator density is highest; e.g., the neutron density (and power) will be highest in the non-boiling regions at the bottom of the core and lowest in the high boiling region at the top of the core.

Some of these are more significant in a BWR and some more significant in a PWR; therefore, the PWR and BWR designs approach the problem of dealing with these causes differently. Examples of the PWR approach to counteract these causes are:

a) Control rods are removed at power by substituting a uniform (soluble) poison and clusters of burnable poison. However, part length control rods may remain in the core to aid in power shaping during steady state and transient conditions.

b) Control rod clusters are used in place of blades and thus the water gaps required for blade insertion are replaced with a series of small separated water holes.

c) and d) The high reactive fuel is loaded near the periphery to offset the leakage effect. This raises the power on the periphery relative to a uniform core and tends to flatten the radial power.

e) The moderator density is essentially uniform (no boiling).

Examples of the BWR approach to counteract these causes are:

a) Most control rods are removed from core region at normal operating power levels by substituting a burnable poison

b) The fuel assemblies are designed such that lower enriched pins are next to the water gaps and since

$$\text{Power } \alpha \ N\sigma_f\phi$$

the high flux near the gap can be offset by the lower U-235 loading.

c) The boiling tends to be greater in the center than on the edge of the core and this helps to flatten power; more reactive assemblies are also loaded near periphery.

d) As implied in (b), higher reactive pins are in the center of the assembly; (see also (c)).

e) The effects of axial variation in moderator density is offset by using bottom entry control rods (which when inserted into the bottom of the core tend to offset the effect of high void in the top of the core); in addition, the axial flux is also shaped by axial positioning of burnable poisons.

As may be apparent, the reduction of peaks and conservatisms by more sophisticated designs and analyses requires more detailed operating information, specifically in-core data, so that the validity of the design is demonstrated; the designer can thus direct his attention to a further reduction in peaks, closing the loop.

1.3 Information Requirements

1.3.1 General Requirements

As stated, the objective of these more sophisticated designs (and here design pertains to fuel placement as well as to fuel assembly design) is to reduce the peaking in the core, which allows the average power density to be increased, within the same operating limits.

Coupled with these advanced designs are the operating strategies, such as control rod placement, which are required to enable the objectives of the designs to be met. Both of these categories;

- design strategies
- operating strategies

require information feedback in order to assure objectives are being attained and to incorporate this experience into future strategies. These information requirements are a function of the time period that is required to react, and can be categorized as follows:

- The core must be monitored more closely to assure that the operating characteristics and hence the margin to limits are as expected. This can be categorized as a short term requirement.

- The information from the monitoring system must be available in a timely fashion so that it can be factored into the operating strategy. This can be categorized as near term requirements.

- The core behavior must be tracked in detail so that it can be ascertained if the design objectives are being met (i.e., are these design considerations really impacting the peaking) and this information factored into future design considerations. This can be categorized as long term requirements.

To meet these information requirements, all LWR vendors have incorporated some type of in-core instrumentation to provide data from which relevant parameters can be inferred. However, these in-core systems are designed and utilized differently in the PWR's and BWR's.

The PWR approach is still similar to that used in the early reactors (Section 1.2). The in-core instrumentation primarily serves an operations support and fuel management function; the short term core monitoring and safety systems functions are still performed by out-of-core instrumentation. (This is discussed in more detail in Section 2.0). However, these out-of-core systems have become more elaborate and are located such that they monitor the differences in behavior in the top and bottom halves of the core as well as behavior differences in the quadrants. The in-core instruments are also used to periodically calibrate these ex-core instruments.

The BWR approach has been to incorporate all safety and monitoring functions as well as the operations support and fuel management functions into the in-core instrumentation. This is due in part to the fact that in a BWR the instruments located external to the core are sensitive to the modes of operation; e.g., control rod motion, flow control, that can change the axial flux distribution (see Section 1.1) and make the relationship between indicated detector core power and actual core power non-linear. Also, with the large cores in current operating reactors, out-of-core instruments do not "see" neutrons that represent a core average, but rather those that are more typical of what is occurring in the region of the core nearest the instrument location and hence may not provide adequate monitoring of the core as an average. (It is questionable whether this is not also the case in a PWR as well.)

In any event, the current BWR's use in-cores for all safety and monitoring while the current PWR's use ex-cores for safety and short term monitoring functions and in-cores for near term (including periodic ex-core instrument calibration) and long term monitoring.

1.3.2 Specific Requirements

With the preceeding as background, let us turn to the specific requirements of modern day plants.

Several parameters have been identified as being necessary for assessing core behavior, either for short term purposes or for near and long term purposes. These parameters are set forth in the tabulations which follow and are presented in terms of a definition, a need, and a requirement. The category of "need" sets the parameter in its operating core context; e.g., short term, near term or long term. The category of "requirement" sets the parameter in its teleological context; i.e., whether it is used to satisfy a regulatory requirement imposed by the NRC or an operational requirement imposed by the utility. The former is classified as a "technical specification" (tech. spec.) and is always delineated in the relevant FSAR. The latter is classified either as an "operation," for short term assessment, or as a "strategy," for near and long term assessment.

In its generic sense, (in-core) power distribution is the measure of the degree of spatial uniformity (or non-uniformity) of energy production. In principle, the power distribution in sufficient spatial and temporal detail provides necessary data to determine if mechanical, thermal, and economic limits are exceeded; or more explicitly if the margins from normal operating conditions to the limits are exceeded.

In practice, the in-core power distributions and all quantities derived therefrom, cannot be measured directly but are inferred from quantities that are directly measurable.

As a composite example of the foregoing, a detector measures the reaction rate in the immediate vicinity of the instrument thimble in which it resides. A set of such detectors, properly placed in the core, can provide a relative U-235 or rhodium reaction rate map at the location of the detectors. The assembly average power in the assemblies adjacent to or including the detector location are deduced by applying factors which are pre-calculated off-site. Local or pin powers are obtained by inference from pin-wise detailed calculations which are also performed off-site. (Section 4 describes the types of analyses and the inherent assumptions used in obtaining these factors.) Experimental and calculational error margins are placed upon these data and parameters pertinent to the information requirements are contructed.

As stated above, the parameters and information generated from the in-core power distribution falls into 3 categories; i.e., short, near and long term requirements. These parameters have been separated into their respective time frames and are given in the following tabulations. The acronym (if there is one), the parameter itself, the definition of the parameter, the need and the requirements for determining it are set forth. Tables 1.1 and 1.2 define the short term parameters for PWR and BWR respectively. Tables 1.3 and 1.4 define the near term parameters for PWR and BWR respectively, and Tables 1.5 and 1.6 define the long term parameters for PWR and BWR respectively.

With these definitions and requirements in mind, Figures 1.3 and 1.4 have been constructed to show this data and information flow and application in more detail so that the reader can see the interactions that occur during this process. Figure 1.1 depicts the BWR data flow and Figure 1.2 depicts the PWR data flow. The 3 PWR systems have been put on the same chart with the flow differences shown when they occur. (These figures have been constructed for comparative purposes and hence may not be exact in every detail.) These two charts start with the sensor output and go through processes and applications that are typical for the reactor type and end with the data and information being sent to the short, near and long term applications for its final disposition.

Figure 1.3 illustrates the eventual use of the information, that is generated per Figures 1.1 and 1.2. This chart shows examples of the information needs and application in the short, near and long term time frames and the interaction among these time frames. This chart is general and is applicable to both BWR's and PWR's.

In the short term, these measured or inferred parameters are compared to their preset limiting values, or margins, and if such margins are exceeded, appropriate action is taken. The appropriate action to be taken covers a broad range; some examples are: power level cutback, shutdown, change of control rod positioning, etc. These actions may be automatic (e.g., scram) or procedural (operator-initiated).

In the near and long term, these inferred parameters are compared to design and operating predictions to determine if the core is operating as expected. If not, appropriate action is taken which would include modifications of re-load patterns, modes of operation, re-load enrichment, operating philosophy, etc.

Hence, these parameters have a direct cause and effect relationship with plant operations and availability. It is at least prudent and perhaps mandatory that these parameters be determined reliably and with a high degree of confidence in the results.

TABLE 1.1

SHORT TERM - PWR

Acronym	Parameter	Definition	Need	Requirement
DNBR	Departure from Nucleate Boiling	The ratio of the local heat flux in an assembly which would cause some point in the assembly to experience departure from nucleate boiling to the actual local heat flux in the assembly.	This is a measure of the margin to a local burnout condition and is continuously monitored.	Tech Spec
A.O.	Axial Offset	Power in the upper half of the core minus the power in the lower half divided by the sum of the two. $\text{A.O.}=\frac{\int^{TOP} P(Z)dZ - \int^{BOTTOM} P(Z)dZ}{\int^{TOP} P(Z)dZ + \int^{BOTTOM} P(Z)dZ}$	This is a measure of the axial skewness of the power distribution and is used to avert gross axial maldistributions and is continuously monitored.	Tech Spec
Q.S.	Quadrant Symmetry	Symmetry of power produced in all four quadrants of the core to within some preset tolerance. $\text{Q.S.} = Q_{P_1}/\frac{1}{4}\sum_{1}^{4} Q_{P_i}$	A lack of quadrant symmetry (called quadrant tilt) implies excess power peaking in one quadrant. This parameter is continuously monitored.	Tech Spec
KW/FT	Kilowatts per Foot	Linear heat generation rate. Often expressed as LHGR.	There is a limit on the KW/FT achievable. Adherence to this limit is monitored thru A.O. and Q.S. monitors and verified periodically by core maps.	Tech Spec
$F_{\Delta H}$	Enthalpy rise hot channel factor	Ratio of the integral of linear power along the rod with the highest integrated power in the reactor to the average rod power.	The peak must be minimized to avoid exceeding local power peak limitations.	Tech Spec
F_q	Total hot channel	This is the peak power in the core relative to the average. $F_q = \frac{\text{MAXIMUM POINT POWER}}{\text{AVERAGE POINT POWER}}$	The peak must be minimized to avoid exceeding local power peak limitations. It is monitored by inference thru the axial offset and quadrant symmetry monitors.	Tech Spec

1 - 10

TABLE 1.1 (Continued)

Acronym	Parameter	Definition	Need	Requirement
F_r^p	Peak pin power in X, Y plane	This is the peak pin power in the X-Y plane relative to the core, or element.	The pin power must be minimized to avoid local power pin locations.	Tech Spec
T_q	Azmuthal Tilt	The power in a quadrant relative to the average of all quadrants is $1 + T_q$ $Q.S. = 1 + T_q$	See Q.S.	Tech Spec
F_r^T	$F_r^P (1 + T_q)$	Peak pin power in symmetrical conditions times the quadrant tilt.	The X, Y pin power must be minimized to avoid local power pin locations.	Tech Spec

TABLE 1.2

SHORT TERM - BWR

Acronym	Parameter	Definition	Need	Requirement
CPR	Critical Power Ratio	The assembly power which causes some point in the assembly to experience transition boiling is defined as the critical power and is a function of axial power shape and recirculation flow. The ratio of the critical power to the actual bundle power is called the critical power ratio.	This parameter is monitored continuously to determine margin to limiting values.	Tech Spec
KW/FT	Kilowatt per Foot	The power produced per foot of pin. (See LHGR)	The maximum KW/FT is monitored continuously to determine margin to limiting value. It is also used to assess present fuel duty relative to the past and regulate the rate of increase of local power allowable.	Tech Spec for maximum value. Administrative procedure for fuel duty assessment.
LHGR	Linear Heat Generation Rate	The power produced per foot of pin for any pin at any axial position and is usually expressed in KW/FT.	LHGR as function of core height must be checked daily.	Tech Spec
CHFR	Critical Heat Flux Ratio	Ratio of the local heat flux to the flux that initiates the onset of the transition boiling in an assembly.	This has been replaced in most BWR's by the critical power ratio (CPR). Formally was continuously monitored to determine margins to limits.	Tech Spec
APLHGR	Average Planar Linear Heat Generation Rate	The maximum power produced per unit height of a fuel assembly. LHGR divided by Local Peaking Factor.	This parameter is monitored continuously to determine margin to limiting value.	Tech Spec

TABLE 1.2 (Continued)

Acronym	Parameter	Definition	Need	Requirement
RBM	Rod Block Monitor	The rod block monitor inhibits control rod withdrawal based on input from the local power range monitors (LPRMs) or minature fission chambers. The trip initiated whenever a variable setpoint is reached.	The rod block monitor prevents abnormally local flux or power peaks from occurring because of rod withdrawal.	Tech Spec
LPF	Local Peaking Factor	The ratio of the peaks pin power in an axial plane of an assembly to the average pin power in the same plane of that assembly.	This factor is used in determining the maximum pin segment power produced in the core.	Tech Spec

TABLE 1.3

NEAR TERM - PWR

Parameter	Definition	Need	Requirement
3-Dimensional Power Distribution	The power at every point in the core relative to the average. Normally given as the average power in each unit volume or fuel segment (node) of the core.	All parameters listed in short term, near term are obtained directly from the power distribution and the long term parameters are a function of what the power distribution was and what it is currently. In fact, all information items are directly or indirectly obtained from the power distribution.	Tech Spec directly and indirectly
Exposure History	The exposure history or exposure distribution is proportional to the total amount of energy produced at each node or fuel assembly segment. The exposure distribution or history is obtained from the integral of the power at each node.	The power distribution is directly dependent on the exposure history or distribution through the reactivity feedback associated with fuel burnup and fission product production.	Tech Spec items are dependent on or derived from the exposure history distribution, e.g., isotopic inventories.
Control Rod Exposure	The control rod exposure is obtained from the total number of neutron that impringe upon the surface of the control rod. It is proportional to and determined from the power history of the nodes adjacent to the control rod while the control rod is inserted.	The control rod exposure is needed to monitor the reactivity worth of the control rod. This is because the poison used in the control rod is consumed with neutron exposure thus decreasing the rods worth.	
Control Rod History	The control rod history is a measure of the period of time a control rod has been adjacent to a fuel segment (node) during the resident life of the fuel assembly. Control rod history is obtained from the integral of the power at each node for the period of time the control rod is present.	The power distribution is directly dependent upon the control rod history through the reactivity feedback. The presence of control rods near a node causes a spectral change and a significant difference in isotopic changes within the fuel element which affect reactivity.	The control rod history is needed to determine the power distribution but is not explicitly required by Tech Specs.

TABLE 1.3 (Continued)

Parameter	Definition	Need	Requirement
Instrument Exposure	The instrument exposure is proportional to the total number of neutrons that impinge upon it e.g., I.E.$\alpha \int n\, dt$. This allows the determination of the depletion of the significant isotope in the instrument.	The instrument exposure is needed in order to analytically correct the output current of the instruments for depletion of the absorbing isotope. This is particularly necessary in the BWR and CE cores where self-powered rhodium detectors are being utilized and where the depletion over the residence life of the detector is significant.	
KW/FT History	The maximum KW/FT that a node (axial segment of a fuel assembly) experiences during a given period of operations. This is a composite array obtained from the power distributions over a period of time.	The KW/FT history is used as a guideline for the local rate of power increases within the core.	Not generally required in current PWR cores. It is anticipated by the author that this item will be logged in the future.
Ex-Core Calibration	The calibration of the ex-core instrumentation such that they can monitor the core performance, by indicating axial offset and quadrant tilt values.	In order to assure that the ex-core detectors are reflecting and monitoring what is actually happening in the core, it is necessary to calibrate them periodically, using in-core detector information.	Tech Specs
Operational Strategies	Operational strategies (in the near term) pertain to the modes of operation utilized in the near future such as power increase or decrease, controlling xenon oscillations by rod movement, etc.	The possible power distributions that can be achieved in the near term (the next hours or days) will be affected to a significant extent upon the maneuvers that are presently being performed with the core.	Procedural

TABLE 1.4

NEAR TERM - BWR

Parameter	Definition	Need	Requirement
3-Dimensional Power Distribution	The power at every point in the core relative to the average. Normally given as the average point in axial planes of the fuel assembly. The peak point in axial planes is also utilized. The power at every point in the core relative to the average. Normally given as the average power in each unit volume or fuel segment (node) of the core.	All items listed in short term, near term are obtained directly from the power distribution and the long term items are a function of what the power distribution was and what it is desired to be. In fact, all information items are directly or indirectly obtained from the power distribution.	Tech Spec directly & indirectly
VOID Distribution	The steam fraction in the coolant water as a function of axial location in each fuel or sub-assembly in the core.	The power distribution is dependent upon the steam-void distribution and vice versa. Therefore, to obtain an accurate power distribution, a representative steam-void distribution is necessary. The steam fraction also enters into the determination of some of the Tech Spec items in the short term table.	Items required by Tech Specs dependent upon the void fraction
KW/ft History	The maximum KW/ft that a node (axial segment of a fuel assembly experiences during a given period of operation. This is a composite array obtained from the power distributions over a period of time.	The KW/ft history is used as a guide-line for the local rate of power increases within the core because of fuel performance requirement.	Procedural
Exposure History	The exposure history or exposure distribution is proportional to the total amount of energy produced at each node or fuel assembly segment. The exposure distribution or history is obtained from the integral of the power at each node.	The power distribution is directly dependent on the exposure history or distribution through the reactivity feedback associated with fuel burnup and fission product production.	Tech Spec items are dependent on or derived from the exposure history distribution, e.g., isotopic inventories-- CPR and MAPLHGR.

TALBE 1.4 (Continued)

NEAR TERM - BWR

Parameter	Definition	Need	Requirement
Control Rod History	The control rod history is a measure of the period of time a control rod has been adjacent to a fuel segment (node) during the resident life of the fuel assembly. Control rod history is obtained from the integral of the power at each node for the period of time the control rod is present.	The power distribution is directly dependent upon the control rod history through the reactivity feedback. The presence of control rods near a node causes a spectral change and a significant difference in isotopic changes which affect reactivity.	The control rod history is needed to determine the power distribution but is not explicitly required by Tech Specs.
Control Rod Exposure	The control rod exposure is obtained from the total number of neutrons that impinge as the surface of the control rod. It is proportional to the power history of the nodes adjacent to the control rod while the control rod is inserted.	The control rod exposure is needed to monitor the reactivity worth of the control rod. This is because the poison used in the control rod is consumed with neutron exposure thus decreasing the rods worth.	
LPRM Calibration	The LPRM's which are used to monitor the core and for deducing the power distribution of the core must be periodically inter-calibrated. This inter-calibration is performed by utilization of the traversing in-core probes which traverse each LPRM string location.	The three dimensional power distribution is inferred from the calibrated LPRM's. Also, the ROD BLOCK MONITOR utilizes the LPRM's signals as input.	A Tech Spec requires that these instruments be calibrated on a monthly basis.
Instrument Exposure	The instrument exposure is proportional to the total number of neutrons that impinge upon it, e.g., i.e., $\int \eta \nu d\tau$	The instrument exposure is needed in order to analytically correct the output current of the LPRM's in the time period between intercalibrations.	The analytical correction of the LPRM output to account for instrument exposure is necessary for reliable power distribution information.

TABLE 1.5

LONG TERM - PWR

Parameter	Definition	Need	Requirement
Refueling Strategies	The refueling strategies are the design and placement of new fuel assemblies in the core. These are a consequence of such items as desired cycle length, fuel burnup objectives, and power shaping considerations. The strategies are directly affected by and related to the past, present, and future power distributions.	Refueling strategies are a necessary part of optimizing core performance and minimizing power peaking. In a PWR fuel placement and orientation is of particular importance because of the large flux gradients at the core periphery.	Procedural
Operational Strategies	Operational strategies (in the long term PWR) pertain to the modes of operation utilized to achieve particular objectives. For example, coast-down to postpone refueling to a more desirable date; partial loading of a core to achieve a desired cycle length, etc.	Optimizing the mode of operation such that the power requirements of the utility are met consistent with the system economics. Load follow is a prime example of this.	Procedural (sometimes needed to remain within operating limits)
Isotopic Inventories	The inventory of elements that are in the strategic material category, such as the plutoniums and uraniums. In each fuel segment or node in the core U-238 and U-235 consumed, P-239, P-240, P-241 and P-242 are created and subsequently consumed. The amount of these elements present are called the isotopic inventories.	The isotopic inventories are a NRC reporting requirement and must be determined on at least an assembly basis. In practice, they can be obtained for every node in the core via the exposure distribution.	Tech Specs
Control Rod Strategies for Operation	The control rod strategies are the placement of the control rods in the core in order to shape the power distribution. The power distribution of the core in the past (through exposure), at present, and in the future are interrelated with the rod strategies.	The power distributions that result from the control rod strategy affect the near term and long term operation of the plant through the exposure histories.	Procedural

TABLE 1.6

LONG TERM - BWR

Parameter	Definition	Need	Requirement
Re-Start Rod Sequence	The re-start rod sequence is the sequence that control rods are withdrawn during a startup of the reactor. This re-start rod sequence is dependent upon past power distributions and the power distribution that is experienced during startup.	The re-start rod sequence is needed to shape the power distribution during a re-start such that operation can remain within such constraints as KW/FT history, rod worth, power to flow limit.	Procedural
Rod Swap Sequence	The rod swap sequence is the sequence of withdrawing and inserting control rods at partial power in order to replace a given set of rods that is in the core by a different set that has not been inserted in the core recently. This sequence also is dependent upon past power distributions (through exposure) and current power distributions.	The rod swap sequence is needed to shape the power distribution during a re-start such that operation can remain within such constraints as KW/ft history, rod worth, power to flow limit.	Procedural
Refueling Strategies	The refueling strategies are the design and placement of new fuel assemblies in the core. These are in consequence of such things as desired cycle time, fuel burnup objectives, and power shaping considerations. The strategies are directly affected and related to the past, present, and future power distributions.	Refueling strategies are a necessary part of optimizing core performance and minimizing power peaking.	Procedural

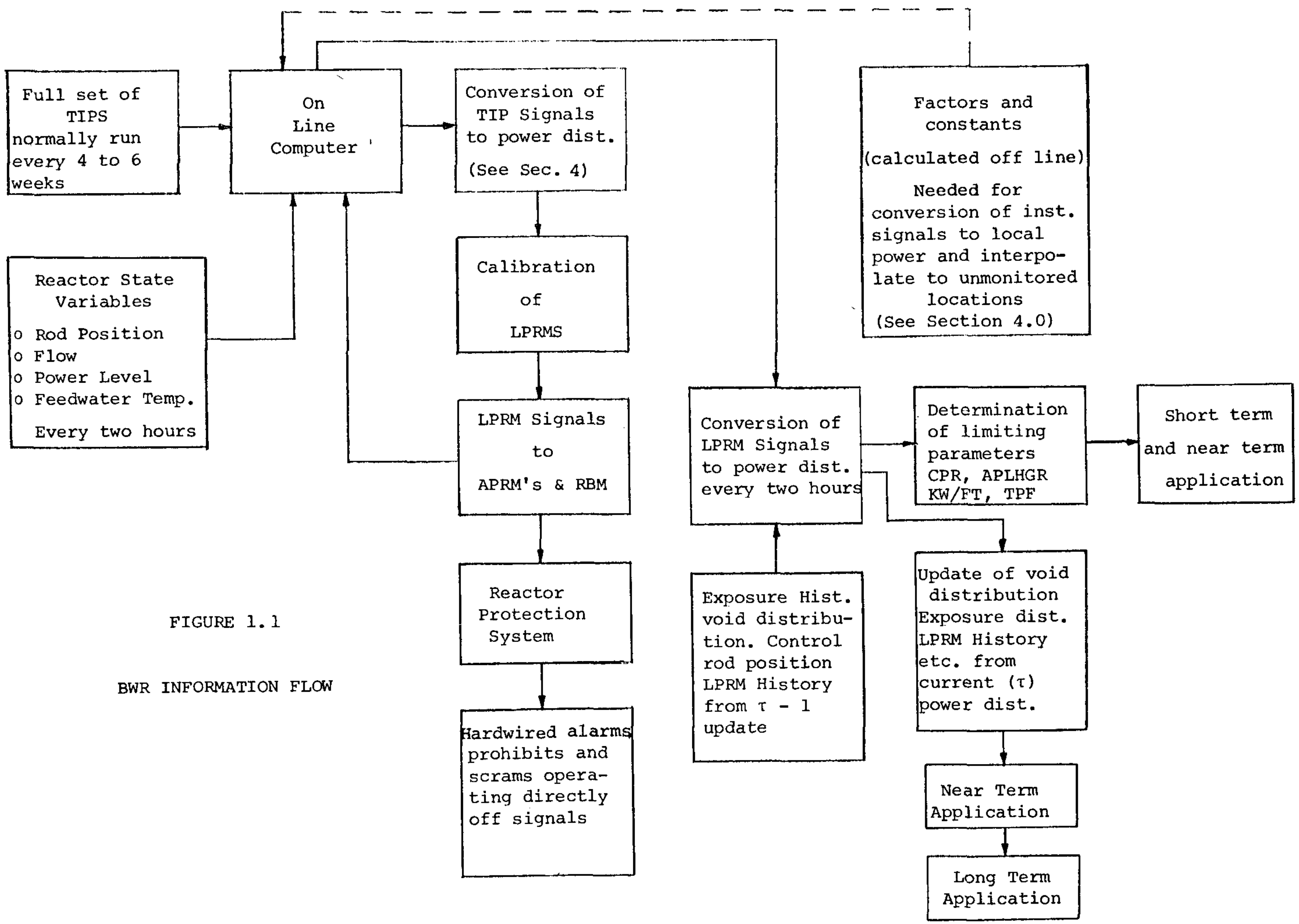


FIGURE 1.1

BWR INFORMATION FLOW

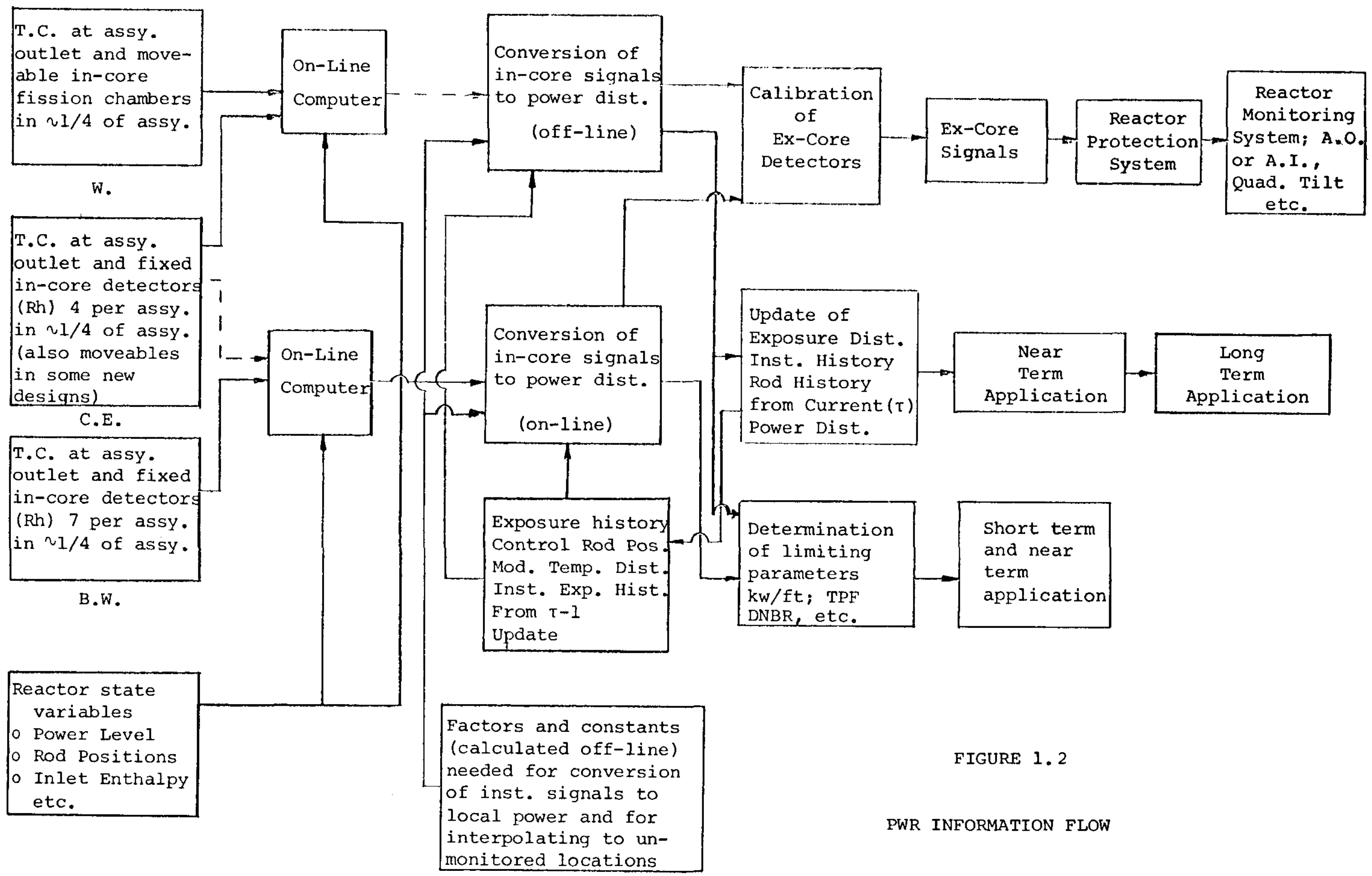


FIGURE 1.2

PWR INFORMATION FLOW

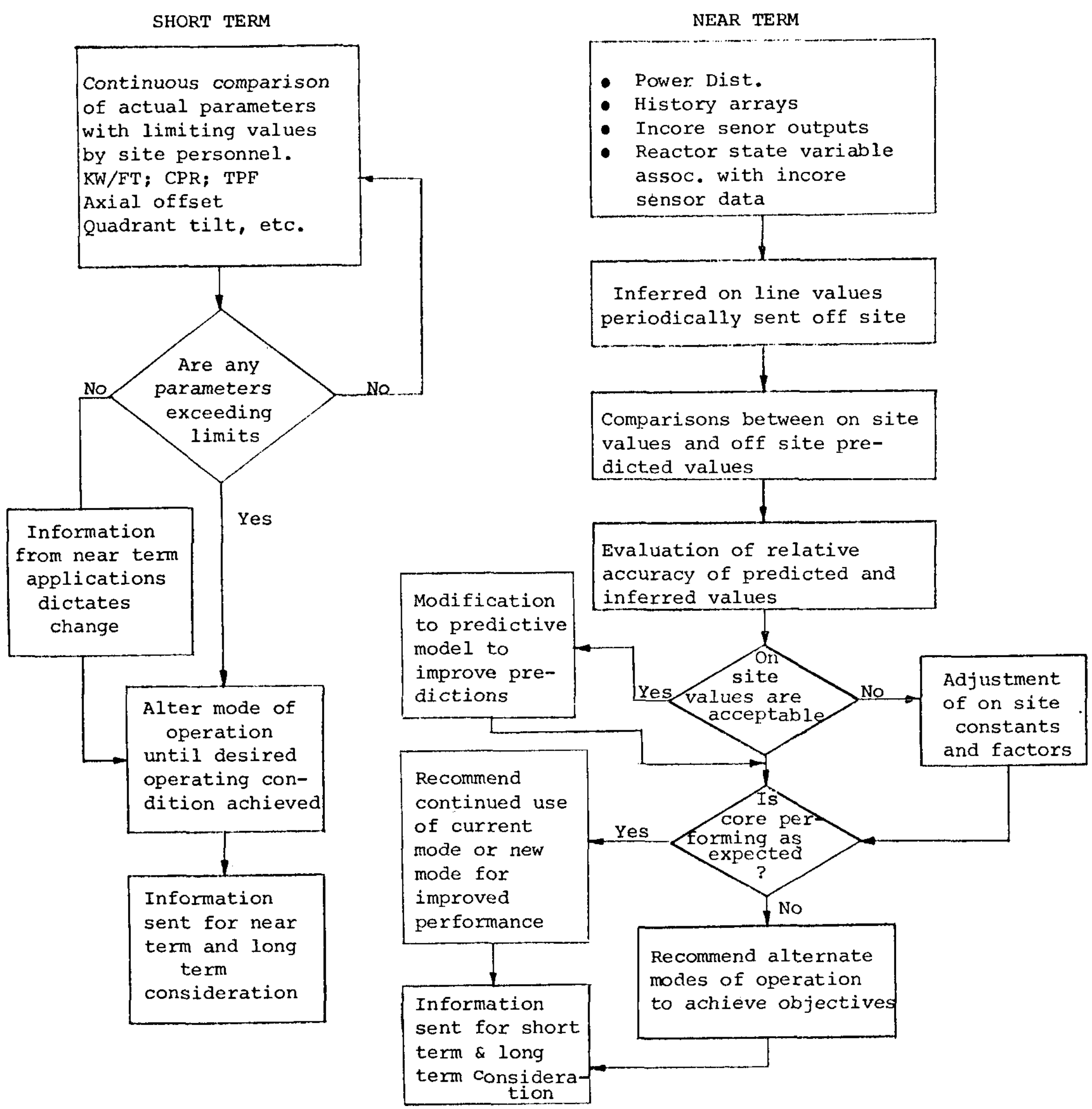


FIGURE 1.3

GENERAL APPLICATION OF INFORMATION

2.0 REVIEW OF THE CURRENT STATE-OF-THE-ART DATA AQUISITION SYSTEMS

This section describes the in-core monitoring systems that are used by each of the light water reactor vendors; Babcock & Wilcox (B&W) Combustion Engineering (C.E.), General Electric (G.E.), and Westinghouse (W).

The objective is to provide enough information on the various systems so that the reader can understand the purpose and intent of the particular system, how the instruments are arranged in the core, the type of instrument and its characteristics, and how often the information from the instruments is actually utilized.

The descriptions of the various systems and their components where taken from vendor material. The sources of the material were: FSAR's, Techinical Specifications, brochures, and other available non-proprietary information.

Each of the LWR manufacturers in-core instrument system's design reflects the philosophy of their approach to monitoring the in-core power distribution. These designs range from movable detectors and no fixed in-core detectors, through fixed in-core detectors and no movable detectors, to fixed in-core detectors and periodic use of movable in-core detectors. Before describing each of the LWR vendors in-core monitoring systems in some detail, it will be helpful to present an overview of some of the similarities and differences of these systems.

Each of the systems use in-core detector material which reacts with neutrons and produces a current proportional to the neutron density at the detector location. Two basic types of detectors are in use, those which use rhodium and are self-power detectors (used by B&W and C.E.*). In both types an analytical procedure is used to convert the power produced in the assembly(s) segment adjacent to the detector. The U-235 detectors must be cross calibrated periodically while the rhodium detectors do not. The U-235 detectors respond immediately to a change

*C.E. also uses some self power detectors of vanadium.

in neutron density while the rhodium detectors have a response time on the order of minutes; i.e., it takes minutes for the detector current to completely reflect the change in neutron density at the detector location. This characteristic eliminates the use of rhodium detectors in any safety systems. The outputs of the rhodium detectors are referenced to the background detector output so that the differential signal is a true measure of the neutron flux.

The in-core detectors can be,

1. movable or traversing in-core detectors
2. fixed in-core detectors.

The movable or traversing detectors are usually U-235 detectors which make an axial traverse at selected locations in the core. The output can be collected by a continuous trace made with an XY plotter and/or by use of a process computer where multiple axial valves are logged. The purposes of the movable in-core detectors are, to provide an axial flux map in selected assemblies from which a core power distribution can be inferred (W does this) and/or to cross calibrate the fixed in-core detectors (G.E. and to a limited degree, B&W and C.E., however, the movable detectors are not a standard part of the B&W and C.E. systems). The local fixed in-core detectors are both U-235 (G.E.) and rhodium (B&W and C.E.). They are placed in selected radial locations encompassing seven axial locations (B&W) or four axial locations (C.E. and G.E.). The Westinghouse system uses no fixed detectors.

The purpose of the fixed in-core detectors is to provide a current which is proportional to the neutron density at the detector locations which can then be converted analytically to the segment power in the assembly(s) adjacent to the detector systems.

As discussed in Section 1.0, there is a basic philosophy difference between BWR's and PWR's. The BWR has only in-core detectors and they provide the signals to the plant protection systems as well as providing the monitoring of the power distribution. The PWR's all use ex-core detectors for the plant protection systems which are periodicallly calibrated from the in-core system. These detectors are located outside the reactor vessel and the basic assumption is that any change in neutron flux level in the core will be accompanied by an equivalent change (percentage wise) in the neutron flux at the detector location. The purpose of the in-core detectors is principally to monitor the core power distribution, however, they are also used for indicating quadrant flux tilts and axial flux imbalances. The ex-cores then are monitoring core power level and power asymmetrics or imbalances. In order to assure they are performing these functions adequately, the ex-cores are periodically (daily) calibrated against a heat balance for indicated core power and periodically (weekly to monthly) against the in-core detectors for flux imbalances and asymmetrics.

The BWR's power monitoring is performed by the in-core detectors. The output of selected in-core detectors, which are located at different axial levels and in different radial locations are used by the power range monitors. The output of the power range monitor goes to the safety system. The power range monitors are calibrated against a heat balance periodically (e.g., twice weekly). The in-core detectors themselves are also cross calibrated periodically (monthly).

Many of the PWR plants have thermo-couples located at the exit of selected fuel assemblies. In theory, one can infer the assembly power distribution from this data; in practice they are used at most, as a secondary system for monitoring "tilt". Table 2.1 also presents some of the information in a comparative manner to aid in understanding the major similarities and differences of these systems.

A more comprehensive flow chart showing the inter-relation of the in-core and the ex-core output and its eventual application was given in Figure 1.2 and Figure 1.2 in Section 1.0.

In the following sections, each vendor system is described separately and each description covers 3 principal areas:

a) System description which includes the system design
    1. System function design objectives
    2. Physical arrangement of the detectors
    3. Type of detectors being used.

IN-CORE SYSTEM COMPARISIONS

TABLE 2.1

	B&W	C.E.	W	G.E.
Detector Type				
Movable	Limited	Limited	Yes	Yes
Fixed	Yes	Yes	No	Yes
Detector Material				
Movable	U-235	U-235	U-235	U-235
Fixed	Rhodium	Rhodium and Vanadium	No Fixed	U-235
Detector Arrangement				
Fixed Axial Locations	7	4	No Fixed	4
% of Core X-Y Locations utilized	∿25%	∿20%	∿25%	∿25%
X-Y Location Relative to Assembly	Middle of Assembly	Middle of Assembly	Middle of Assembly	Corner of Assembly
Intercalibration	No	Limited	Yes—	Yes
On Line Interpretation Capability	Yes	Limited	No	Yes
Need Off Line Interpretation	No	Yes	Yes	No
Use in Plant Protective System	Calibration	Calibration	Calibration	Directly
Plant Protective System Signals Detector Location	Out of Core	Out of Core	Out of Core	In Core

b) Sampling rates which describes the frequency at which the in-core power distribution and/or related quantities are determined from the in-core instruments.

c) Neutron Instrumentation Uncertainties which describes the accuracy of the in-core detectors as derived of the vendors.

2.1 General Electric Reactors

Figure 2.1.1 is a schematic showing the information flow and interaction among the various components in the G.E. system. These components and their functions are described in the following section. These descriptions have been taken principally from References 1 and 2.

2.1.1 System Description

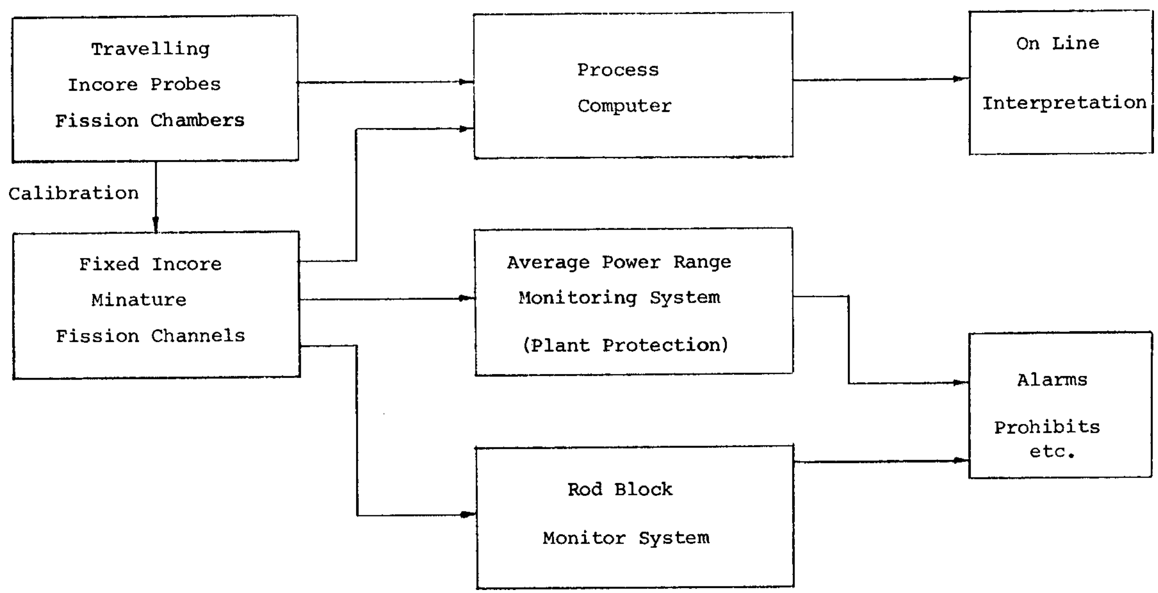
2.1.1.1 Fixed In-Core Detectors

A. System Design:

1) Local Power Range Monitors:

The design functions of the Local Power Range Monitors (LPRMs) are as follows:

- The LPRMs provide signals proportional to the local neutron flux at various locations within the reactor core to the Average Power Range Monitor Subsystem (APRMs), so that accurate measurements of average reactor power can be made.
- The LPRMs supply signals to the Rod Block Monitor Subsystem, so that measurement of changes in local relative neutron flux can be made during the movement of control rods.
- The LPRMs are capable of alarming under conditions of high or low local neutron flux indication.
- The LPRMs supply signals proportional to the local neutron flux to the process computer to be used in power distribution calculations, local heat flux calculations, minimum critical heat flux calculations, and fuel burnup calculations, etc.
- The LPRMs supply signals proportional to the local neutron flux to drive indicating meters and auxiliary devices to be used for operator evaluation of the power distribution, local heat flux, minimum critical heat flux, and fuel burnup.



G.E.

FIGURE 2.1.1

Block Diagram of Typical General Electric Power Distribution Monitoring System

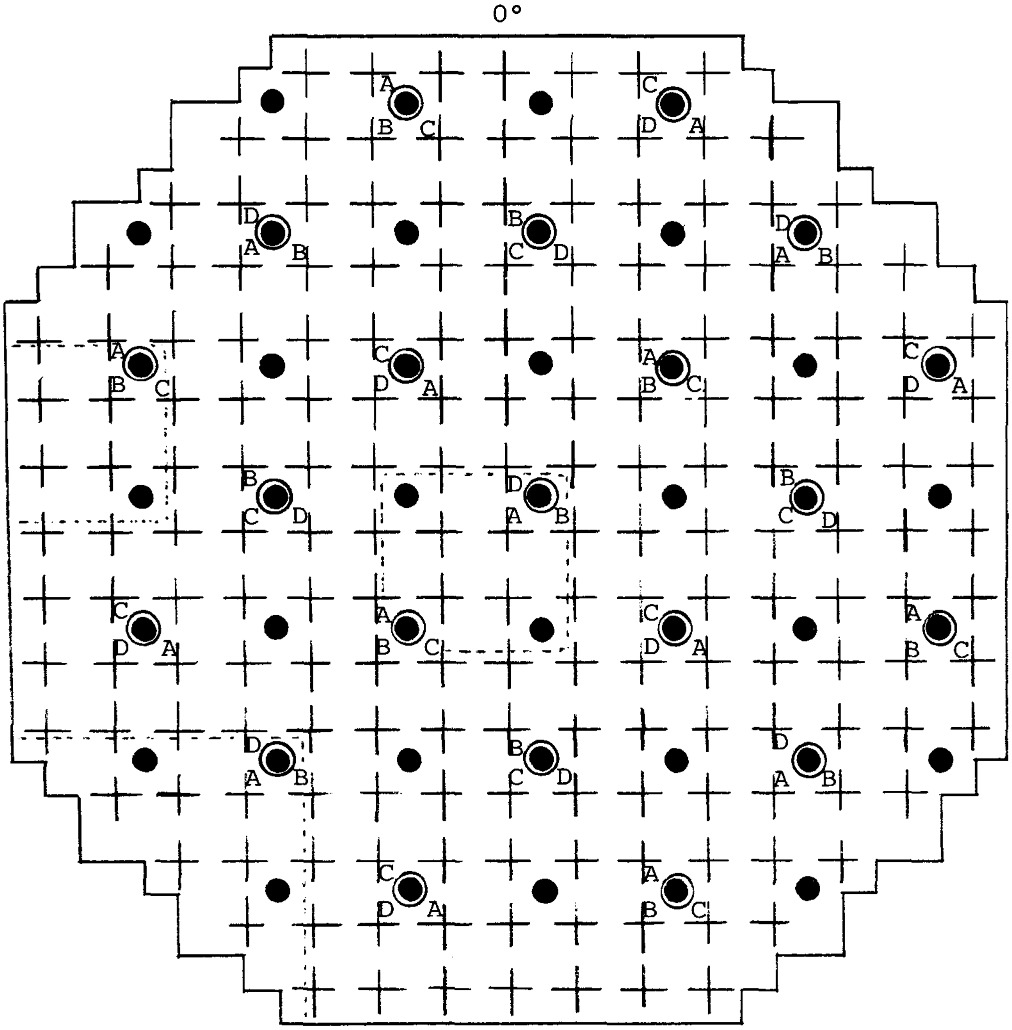
The Local Power Range Monitor Subsystem, as calibrated by the Traversing In-Core Probe Subsystem, provides detailed information about the neutron flux throughout the reactor core. The division of the LPRMs into various groups for dc power supply allows operation with one dc power supply failed or being serviced without limiting reactor operation. Individual failed chambers can be bypassed, and neutron flux information for a failed chamber location can be interpolated from nearby chambers. A substitute reading for a failed chamber can be derived from an octant-symmetric chamber, or an actual flux indication can be obtained by insertion of a TIP to the failed chamber position.

2) Average Power Range Monitor Subsystem Design:

- The design of the APRMs shall be such that for the worst permitted input LPRM bypass conditions, the APRMs shall be capable of generating a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage.
- The design of the APRMs shall be consistent with the requirements of the safety design basis of the Reactor Protection System.
- The APRMs shall provide a continous indication of average reactor power from a few percent to 125 percent of rated reactor power.
- The APRMs shall be capable of providing trip signals for blocking rod withdrawal when the average reactor power exceeds pre-established limits set to prevent scram actuation.
- The APRMs shall provide a reference power level for use in the Rod Block Monitor Subsystem.

The APRM channels use input signals from a number of LPRM channels. In a 1000 MW type reactor three APRM channels are associated with each of the trip systems of the Reactor Protection System. The LPRMs that make up each APRM are scattered axially and radially so that the APRM input is representative of a core average (see Fig. 2.1.2).

CORE TOP VIEW



LPRM DETECTOR CODE

D = Top Detector in String

C = Upper Middle Detector

B = Lower Middle Detector

A = Bottom Detector in String

APRM/LPRM ASSIGNMENT CODE

⊙ LPRM Input to APRM Channels A, C, E for Reactor Protection Channel A

Upper Left Letter is LPRM Assigned to APRM A

Lower Left Letter is LPRM Assigned to APRM C

Lower Right Letter is LPRM Assigned to APRM E

Dashed Lines Indicate Control Rods Associated with a Given Rod Block Monitor (RBM). The LPRM's Within the Dashed Lines Feed the Specific RBM.

FIGURE 2.1.2

LPRM LOCATION IN 1000 MWE GE-BWR

3) Rod Block Monitor Subsystem Design:

The RBMS has two RBM channels each of which uses input signals from a number of LPRM channels. A trip from either RBM channel can initiate a rod block. One RBM channel may be bypassed without loss of subsystem function.

- The RBMS is designed to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass.
- The RBMS provides a signal to permit operator evaluation of the change in the local relative power level during control rod movement.

The RBM signal is generated by averaging a set of LPRM signals. One of the RBM's averages the signals from LPRM detectors at the A and C positions (Fig. 2.1.3) in the assigned power range detector assemblies, and the second RBM averages the signals from the LPRM detectors at the B and D positions in the assigned power range detector assemblies. Assignment of power range detector assemblies to be used in RBM averaging is controlled by the selection of control rods. The RBM is automatically bypassed and the output set to zero if a peripheral control rod is selected. If any LPRM detector assigned to a RBM is bypassed, the computed average signal is adjusted automatically to compensate for the number of LPRM input signals to average.

The magnitude of each RBM channel output is normalized to an assigned APRM channel whenever a control rod is selected. A signal from one APRM channel assigned to each reactor protection system trip system supplies this reference signal for the RBM channel on that same trip system. This gain setting is held constant during the movement of that particular control rod, thus providing an indication of the change in the relative, local power level. If the APRM used to normalize the RBM reading is

FIGURE 2.1.3

IN-CORE NEUTRON DETECTOR ASSEMBLY AXIAL ARRANGEMENTS

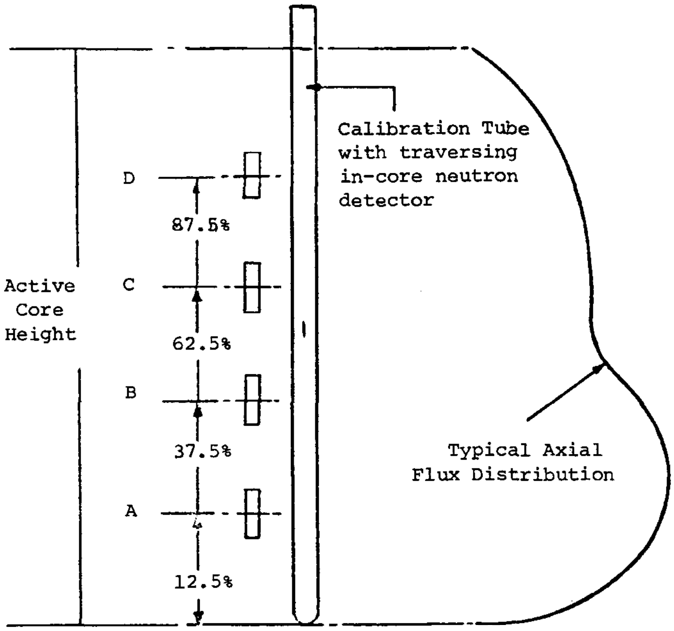
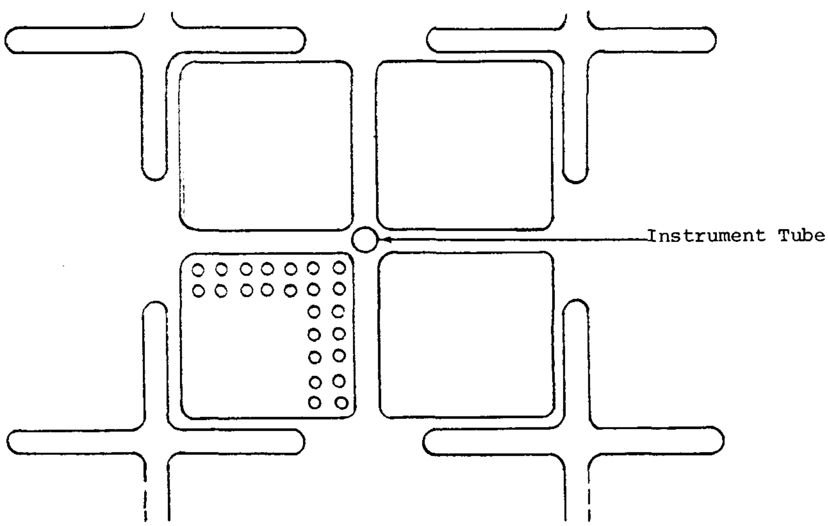


FIGURE 2.1.4

INSTRUMENT TUBE LOCATION



indicating less than 30 percent power, the RBM is zeroed and the RBM outputs are bypassed. If the normalizing APRM is bypassed, the normalizing signal is automatically provided from a second RBM. In the operating range the RBM signal is accurate to about 1 percent of full scale of the correct signal including all variances due to drift, environmental changes (normal control room variations), and supply voltage variations.

The RBM supplies a trip signal to the reactor manual control system to inhibit control rod withdrawal. The trip is initiated whenever the RBM output exceeds the rod block set point. There are three parallel rod block set point lines which have an adjustable slope. These set point lines provide a set point that is a function of the recirculation driving loop flow. The intercepts of these set point lines with rated flow are adjustable. The normal settings will be approximately 108 percent for the upper line, 98 percent for the intermediate line, and 88 percent for the lower line. Lights indicate which rod block set point line is active. Two percent below the intermediate and lower rod block set point are the set up permissive and set down lines. These lines on increasing power will light a set up permissive indicator so that the operator can evaluate the conditions and manually change to the next higher rod block set point line. On decreasing power these lines will provide automatic set down. One of the two RBMs can be bypassed at any time by operator action. Either RBM can inhibit control rod withdrawal.

B. Physical Arrangement:

The LPRM system includes LPRM detectors throughout the core at different axial heights. Figure 2.1.2 illustrates the LPRM detector radial layout scheme which positions a detector assembly at every fourth intersection of the narrower of the water channels around the fuel bundles (narrow-narrow water gap). Thus, every narrow-narrow water gap, except peripheral assemblies, has either an actual detector assembly or a symmetrical equivalent assembly in some other quadrant.

In each of the in-core instrument assembly locations shown in Figure 2.1.2 there are four LPRMs which are fixed axially. Figure 2.1.2 shows the axial location of the LPRMs in a given string. In addition, as described earlier, each in-core instrument assembly also accomodates a Traveling In-Core Probe (TIP) for cross calibrating LPRMs.

Figure 2.1.4 shows a four assembly cluster about an instrument tube. The control blade locations relative to the instrument tube is also depicted. Unlike the PWR's, the BWR thimble is located external to any fuel assembly and hence is affected by conditions in all four adjacent assemblies.

As mentioned, the LPRMs which are associated with a APRM are scattered throughout the core. Figure 2.1.2 shows a typical distribution of LPRMs assigned to APRM channels. (The LPRMs for 3 APRM channels are represented. The remaining LPRM strings are associated with three other APRM channels).

The minimum number of LPRM inputs required for each RBM channel to prevent an instrument inoperative alarm is 4 when using 4 LPRM assemblies, 3 when using 3 LPRM assemblies and 2 when using 2 LPRM assemblies (see Figure 2.1.2).

As shown in Figure 2.1.2 the number of LPRMs which feed an RBM channel depends upon the location of the rod that is being withdrawn and the number of available LPRMs in the immediate vicinity. For example, a rod near the periphery may have only two LPRMs assemblies that "see" the local neutron population change as the rod is being withdrawn.

C. Type of Detector:

The LPRMs consist of the fission chamber detectors, the signal conditioning equipment, and trip functions. The LPRM signals are used in the APRMs (Section 2.1.1.2), RBMs (Section 2.1.1.3), and Process Computer.

The LPRM detector assemblies, each containing four fission chambers, are distributed to monitor four horizontal planes throughout the core. The detector assemblies are inserted into the core in spaces between the fuel assemblies through thimbles which are mounted permanently at the bottom of the core lattice and which penetrate the bottom of the reactor vessel.

Each LPRM detector assembly contains four minature fission chambers with an associated solid sheath cable. Each fission chamber produces a current which when coupled with the LPRM signal conditioning equipment provides the desired scale deflection throughout the design lifetime of the chamber. Each individual chamber of the assembly is a moisture-proof, pressure-sealed unit. Each assembly also contains a calibration tube for a Traversing In-Core Probe (TIP). The enclosing tube around the entire assembly contains holes evenly spaced along its length. These holes allow circulation of the reactor coolant water to cool the fission chambers. According to G.E., numerous tests have been performed on the chamber assemblies including tests of linearity, lifetime, gamma sensitivity, and cable effects.

The four minature fission chambers used on each assembly are designed to operate up to a temperature of 559 F and a pressure of 1250 psig. The chambers are vertically spaced in the LPRM detector assemblies in such a manner as to give adequate axial coverage of the core, complementing the radial coverage given by the horizontal arrangement of the LPRM detector assemblies. Each minature chamber consists of two concentric cylinders, which act as electrodes. The inner cylinder, the collector, is mounted on insulators and is separated from the outer cylinder by a small gap. The gas between the electrodes is ionized by the charged particles produced as a result of neutron fissioning of the uranium coated outer electrode. The chamber has at the beginning of operation, a sensitivity of approximately 2.15×10^{-17} amps/nv and is operated at a polarizing potential of approximately 150V. The negative ions produced in the gas are accelerated to the collector by the potential difference maintained between the electrodes. In a given neutron flux, all the ions produced in the ion chamber can be collected if the polarizing voltage is high enough. When this situation exists, the ion chamber is considered to be saturated. Output current is then independent of operating voltage and has a linearity of 1 percent (1σ) over the design operating range.

2.1.1.2 <u>Traversing In-Core Probe Subsystem (TIPS)</u>

A. System Design:

The design objectives of the TIP system are as follows:

- The TIPs are capable, as they traverse, of providing a signal proportional to the axial neutron flux distribution at selected small axial intervals over the regions of the core where LPRM detector assemblies are located. This signal has sufficient precision to allow reliable calibration of LPRM gains.

- The TIPs positioning machinery shall provide accurate indication of the position of the flux measurement to allow pointwise or continuous measurement of the axial neutron flux distribution.

B. Physical Arrangement:

TIPs can traverse LPRM locations which are shown in Figure 2.1.2.

C. Type of Detector:

In a typical 1000 MWE plant the TIPs include five traversing in-core probe (TIP) machines each of which has the following components: (Figure 2.1.5)

- 1 traversing in-core probe (TIP)
- 1 drive mechanism
- 1 indexing mechanism
- Up to 10 in-core guide tubes
- 1 chamber shield

The subsystem allows calibration of LPRM signals by correlating TIP signals to LPRM signals as the TIP is positioned in various radial and axial locations in the core. The guide tubes inside the reactor are divided into groups. Each group has its own associated TIP machine.

A TIP drive mechanism uses a fission chamber attached to a flexible drive cable, which is driven from outside the primary containment by a gear box assembly. The flexible cable is contained by guide tubes that continue into the reactor core. The guide tubes are a part of the LPRM detector assembly and are specially prepared to provide a durable low friction surface. The indexing mechanism allows the use of a single detector in any one of ten different tube paths. The tenth tube is used for TIP cross calibration with the other TIP machines. The control system provides both manual and semi-automatic operation. The TIP signal is amplified and displayed on a meter. Core position versus neutron flux is recorded in the main control room on an X-Y recorder.

The heart of each TIP machine is the probe (Figure 2.1.6) consisting of a detector and the associated signal drive

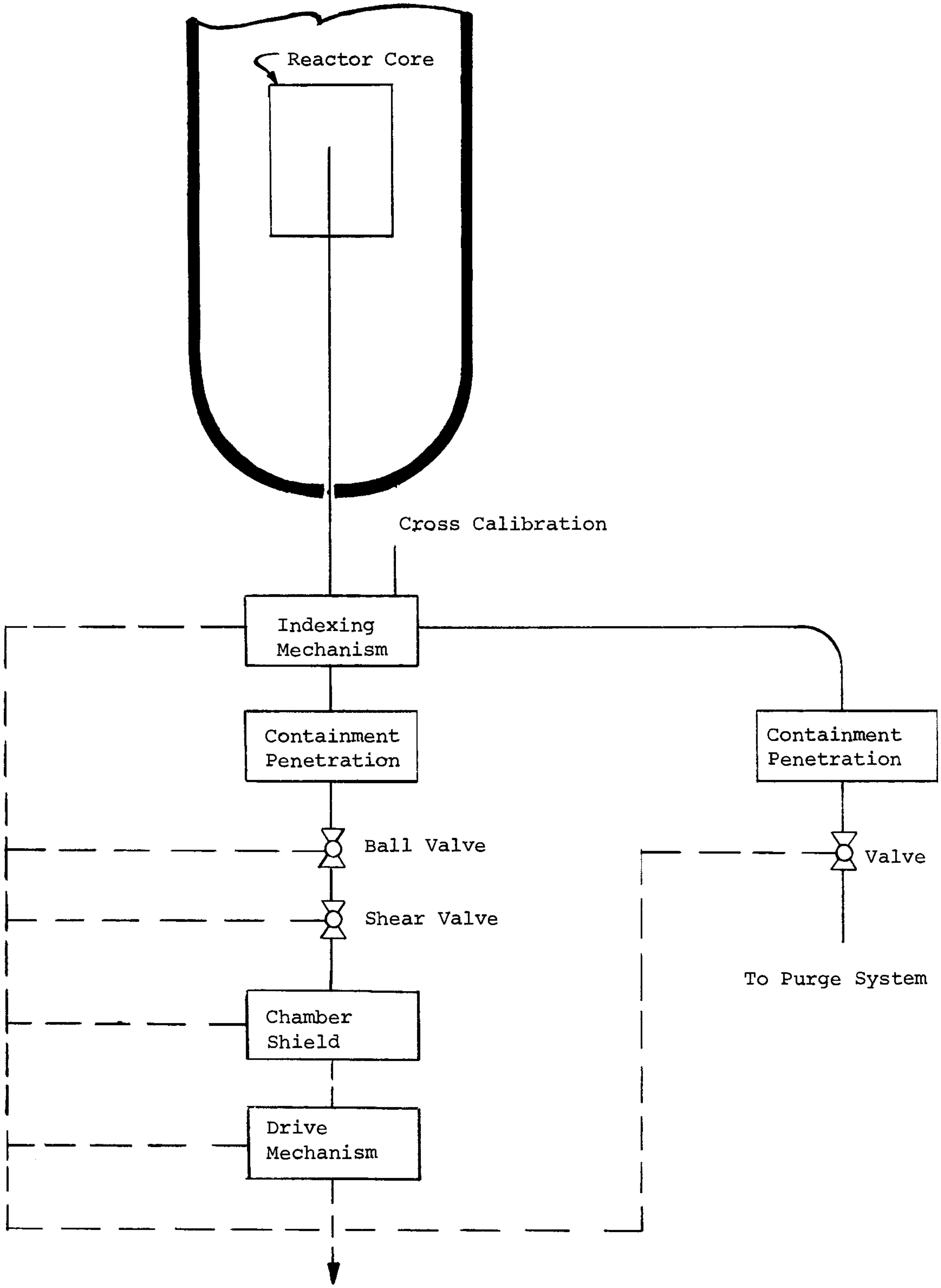


FIGURE 2.1.5

Typical GE-BWR Traveling Incore Probe Configuration

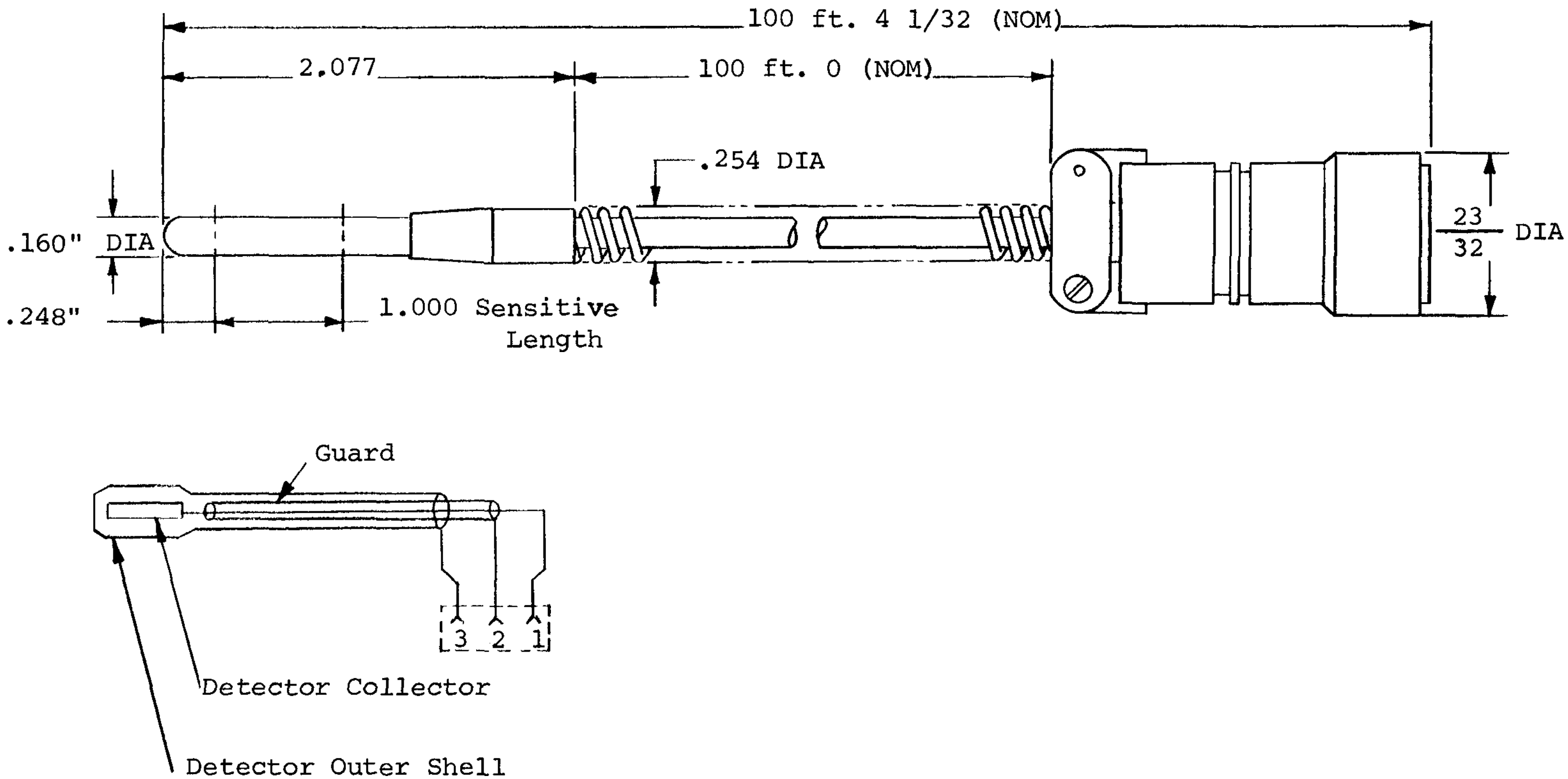


FIGURE 2.1.6

TRAVERSING INCORE PROBE ASSEMBLY

of the cable drive motor. The analog position signal from a potentiometer and a flux amplifier output are used to plot neutron flux versus in-core position of the TIP. The TIP position signal is also available to the process computer. The digital counter is used to position the TIP in the guide tube through the control logic with a linear position accuracy of plus or minus one inch. The digital counter can control TIP positions at the top of the core for initiation of scan, and at the bottom of the core for changing to fast withdrawal speed.

2.1.2 Sampling Rates

The LPRM signals are monitored on a continuous basis and as stated earlier (and shown in Figure 2.1.2) they feed the APRM and RBM monitor systems which are part of the reactor protection system. The APRM system is calibrated twice a week using a heat balance.

The use of these signals for the generation of the in-core power distribution and the parameters derived therefrom, is done automatically on a periodic basis (every 2 hours) or upon demand. The peak heat flux is checked at least once per day; i.e., TPF is determined daily.

The LPRMs are inter-calibrated by the TIPs periodically ($\sim$1 month intervals) by performing a full core map. In addition, the LPRMs are re-calibrated after significant power or rod pattern changes. This is accomplished by a full core map or a single LPRM string traverse depending upon the severity of the changes.

2.1.3 Neutron Instrumentation Uncertainties

General Electric has performed a study on the accuracy of the process computer[3]. The following uncertainties in the TIP and LPRM signals are taken from this report.

The TIP signal uncertainty arises from both the geometric mislocation of the TIP detector and the neighboring fuel channels with respect to their nominal design positions and the random neutron, electronic, and boiling noise in the reactor. The random noise component of the signal has been determined by traversing a common instrument tube with the detector and recording the variance in the signal. The resulting random noise uncertainity in the TIP data (all values are with respect to a 6-inch segment) was found to be 1.2%. The geometrical component has been determined by comparing the random deviation from unity of the ratio of symmetrically located TIP signals (which should have identical readings during symmetric operation). This random deviation is the statistical superposition of the TIP geometrical and random noise uncertainties. The geometrical uncertainty, determined by statistically subtracting the known random noise component from the total observed deviation, was determined to be 2.3%. The geometrical and random noise TIP uncertainties combine to give an overall TIP uncertainty of 2.6%.

The local power range instruments are used in the core performance evaluation to update the base TIP axial flux profiles after small power changes ($\leq$ 15% of rated). The LPRM instrument signal uncertainty arises from the axial interpolation of the signals, random signal noise, system nonlinearity, and instrument sensitivity decay. LPRM-geometrical uncertainties are normalized out of the power evaluation. Estimates of these individual component uncertainties have been made based on the instrument design, a 15% (of rated) change in power and the maximum period between LPRM calibrations - 30 days - and lead to an overall uncertainty of 3.4% for the LPRM update to be applied to the base TIP data. This LPRM update uncertainty when combined with the base TIP uncertainty of 2.6% yields a resultant 4.3% uncertainty in the LPRM-extrapolated TIP signals.

These results are for initial cores, and G.E. states that more recent studies indicate an uncertainty of ∿5.0% in reload core due to mislocation of TIP flux detectors.

Based on the authors experience however, it should be pointed out that in some of the operating BWRs the TIP variations based on symmetrically located TIPs is significantly larger than 2.6% or the 5%.

2.2 Westinghouse Reactors

Figure 2.2.1 illustrates the interrelation and interaction of the major components in the Westinghouse system for monitoring the in-core power distribution. Much of the following descriptions were taken from References 4 and 5.

2.2.1 System Description

2.2.1.1 In-Core Detectors

A. System Design:

The in-core instrumentation is designed to yield information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. Using the information thus obtained, it is possible to confirm the reactor core design parameters. The system provides means for acquiring data only, and performs no operational plant control.

The in-core instrumentation system consists of thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected locations; and flux thimbles, which run the length of selected fuel assemblies and accommodate a fission chamber which traverses the length of the assembly. A typical design calls for 39 thermocouples and 36 flux thimbles.

The movable fission chambers produce a current proportional to the fission rate in the chamber. As they traverse the selected fuel assembles, the detector output current provides a measure of the relative neutron density in the instrument tube. The relative current is then converted analytically (section 4.0) to the relative fission rate or power distribution in the surrounding assembly and subsequently extrapolated to the assemblies that do not accommodate a detector.

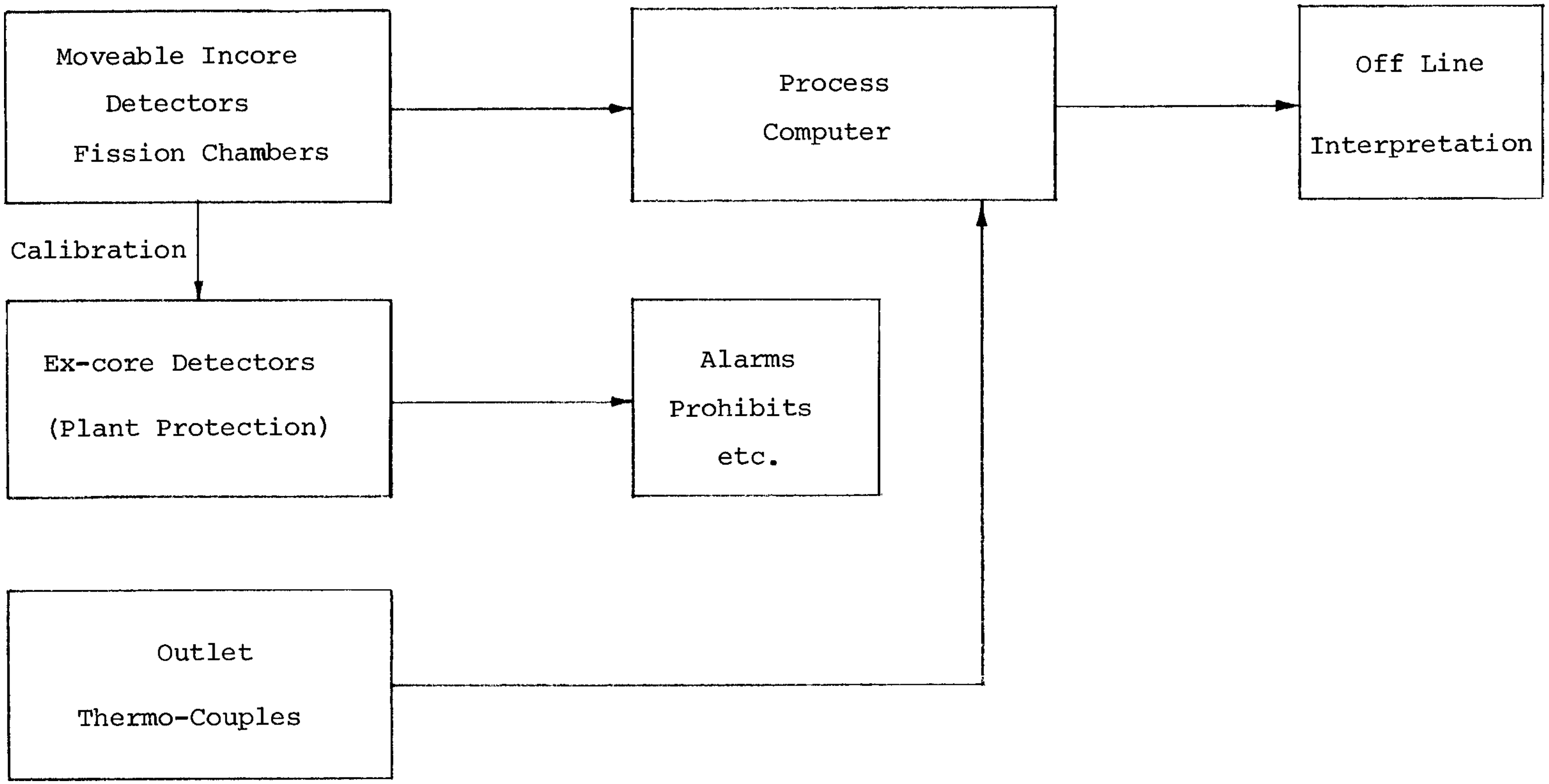


FIGURE 2.2.1

Block Diagram of Typical Westinghouse Power Distribution Monitoring System

This resulting in-core power distribution is used to determine the various parameters of interest (Table 1.1, 1.3 and 1.5). The thermocouples produced a current proportional to the moderator temperature at the outlet of selected assemblies. Knowing the inlet temperature, a ΔT can be determined for these assemblies and the total power produced in these assemblies can be inferred.

The data obtained from the in-core temperature and flux distribution instrumentation system, in conjunction with previously determined analytical information, can be used to determine the fission power distribution in the core at any time throughout core life. The in-core instrumentation provides information which may be used to calculate the coolant enthalpy distribution: the fuel burnup distribution; and to estimate the coolant flow distribution. This method is more accurate than using calculational techniques alone.

Both radial and azimuthal symmetry of power distributions may be evaluated by comparing the detector and thermocouple information from one quadrant with similar data obtained from the other three quadrants.

Thermocouple readings are monitored by the computer with backup readout provided by a precision indicator with manual point selection. Information from the instrumentation is available even if the computer is not in service.

The miniature neutron flux detectors, suitable for being remotely positioned in the core, provide remote readout for flux mapping. Retractable thimbles, into which the miniature detectors are driven, are pushed into the reactor core through conduits that extend from the bottom of the reactor vessel down through the concrete shield area, then to a thimble seal table.

The control room contains the necessary equipment for control, position indication and flux recording. Panels are provided to indicate the position of the detectors, and for plotting the flux level versus the detector position. Additional panels are provided for such features as drive motor controls, core path selector switches, plotting and gain controls. A "flux-mapping" operation consists of selecting (by panel switches) flux thimbles at various core locations. The detectors are driven to the top of the core and stopped automatically. An X-Y plot from each detector (position vs. flux level) is initiated with the slow withdrawal of the detectors through the core from top to a point below the bottom. All four detectors or any combination of them may be used simultaneously for flux plotting. In a similar manner, other core locations are selected and plotted.

Each detector provides axial flux distribution data along the center of a fuel assembly. Various radial positions of detectors are then compared to obtain a flux map for a region of the core.

B. Physical Arrangement:

The thimbles are distributed nearly uniformly over the core, with about the same number of thimbles in each quadrant. The number and location of these thimbles have been chosen to permit measurement of local to average peaking factors to an accuracy of ±10 percent (95 percent confidence) (Ref. 5). Measured nuclear peaking factors are then increased by 10 percent to allow for possible instrument error. The DNB ratio calculated with the measured hot channel factor is then compared to the DNB ratio calculated from the design nuclear hot channel factors. If the measured power peaking is larger than expected, power reduction must be initiated. Figure 2.2.2 shows the instrument assembly location.

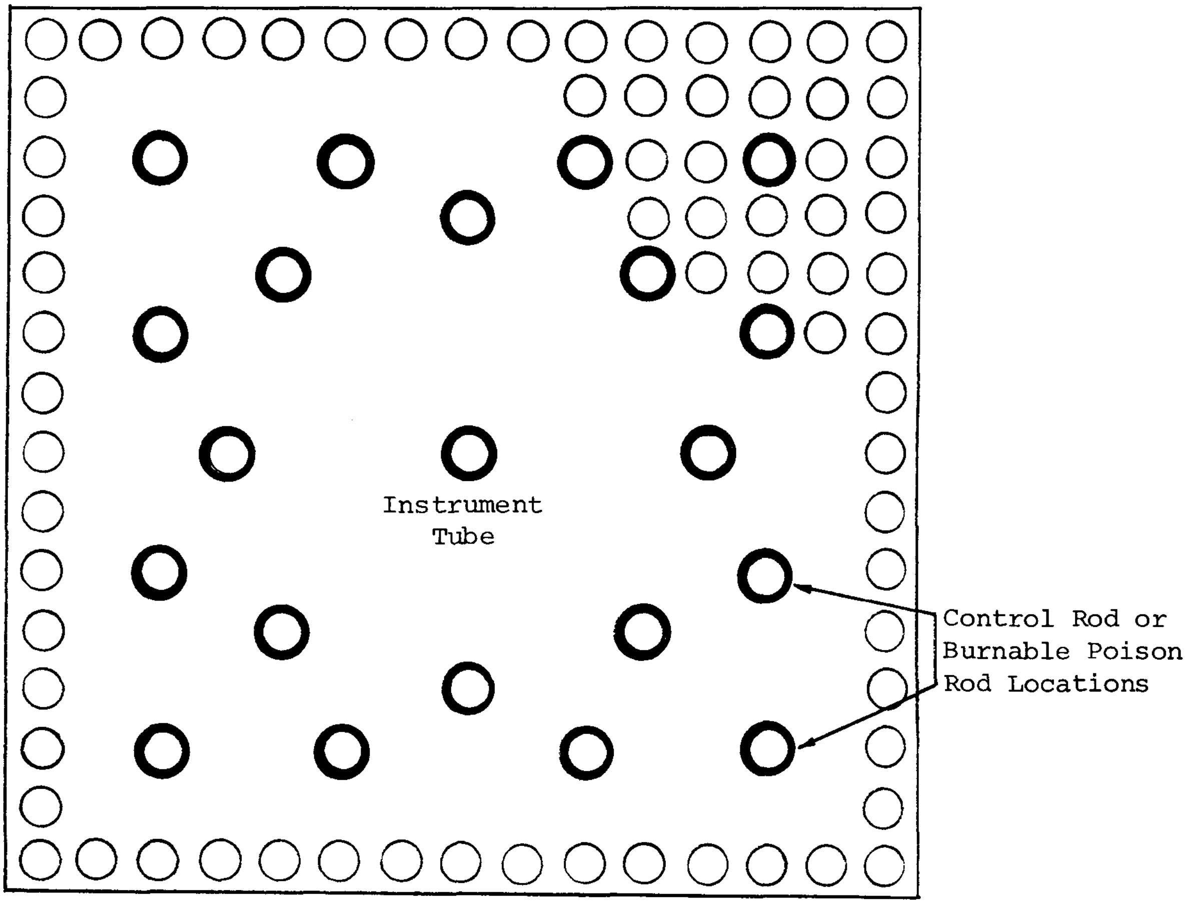


FIGURE 2.2.2

FUEL ASSEMBLY AND INSTRUMENT TUBE LOCATION

See Figures 2.2.3 and 2.2.4 for the locations of the flux thimbles and the thermocouples for various size plants. As can be seen, some of the thermocouple locations and the movable in-core detector (MIDs) location are azimuthally symmetric. In addition, each movable in-core detector can access more than one fuel assembly so that they can be cross calibrated before (and during) an in-core flux map. Table 2.2.1 summarizes the numbers of MIDs and thermocouples in 2, 3 and 4 loop plants.

TABLE 2.2.1

	Thermocouples	Flux Thimbles	Number of Movable In-Core Detectors
2 loop	39	36	4
3 loop	51	50	5
4 loop	65	58	6

C. Type of Detector:

The in-core instrumentation system consists of cromel-alumel thermocouples at the output of specific fuel assemblies and movable neutron detectors which can be positioned anywhere along the vertical axis of selected fuel assemblies. A position indicator in the control room provides the operator with indication of the location of the probe within an accuracy of one inch. Figure 2.2.5 depicts the drive mechanism.

Stainless steel encapsulated neutron detectors are driven into one of the thimbles. These are U-235 fission chamber neutron detectors of dimensions:

0.188 in. diameter x 2.1 in. length

Switches within each transfer device provide feedback of the proper path selection.

3 LOOP PLANT

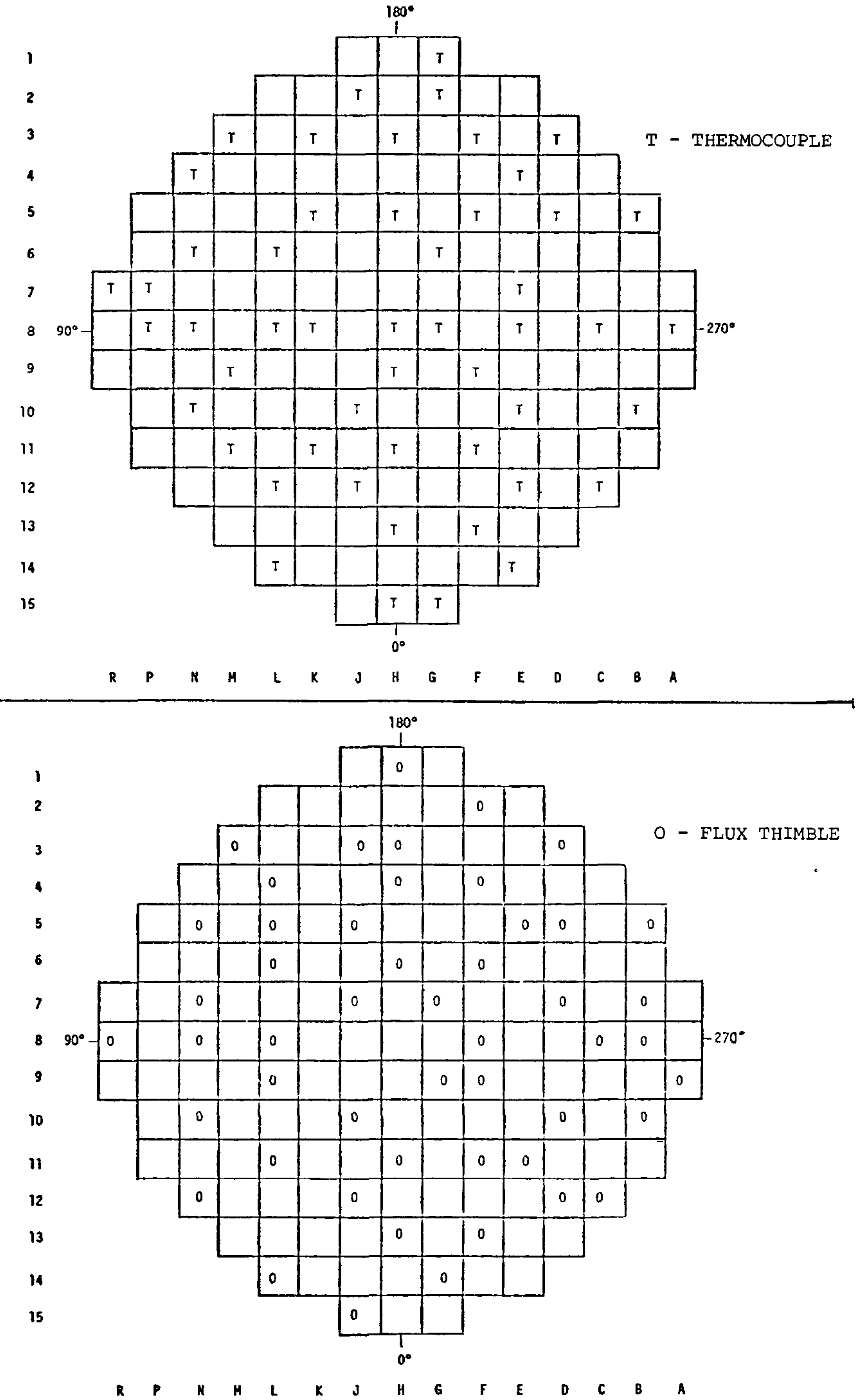


FIGURE 2.2.3

DISTRIBUTION OF THERMOCOUPLES AND FLUX THIMBLES

4 LOOP PLANT, TYPE 2

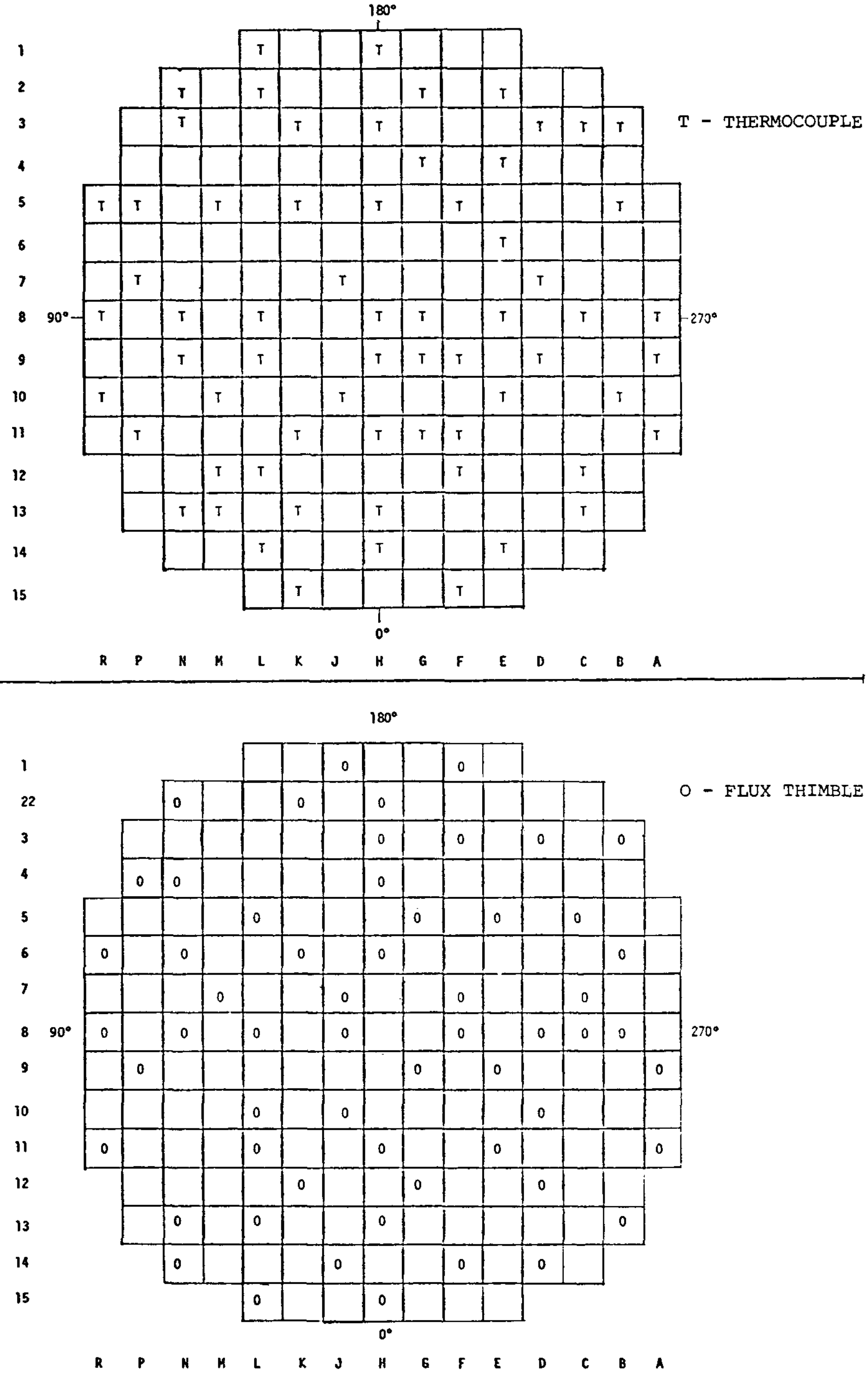


FIGURE 2.2.4

DISTRIBUTION OF THERMOCOUPLES AND FLUX THIMBLES

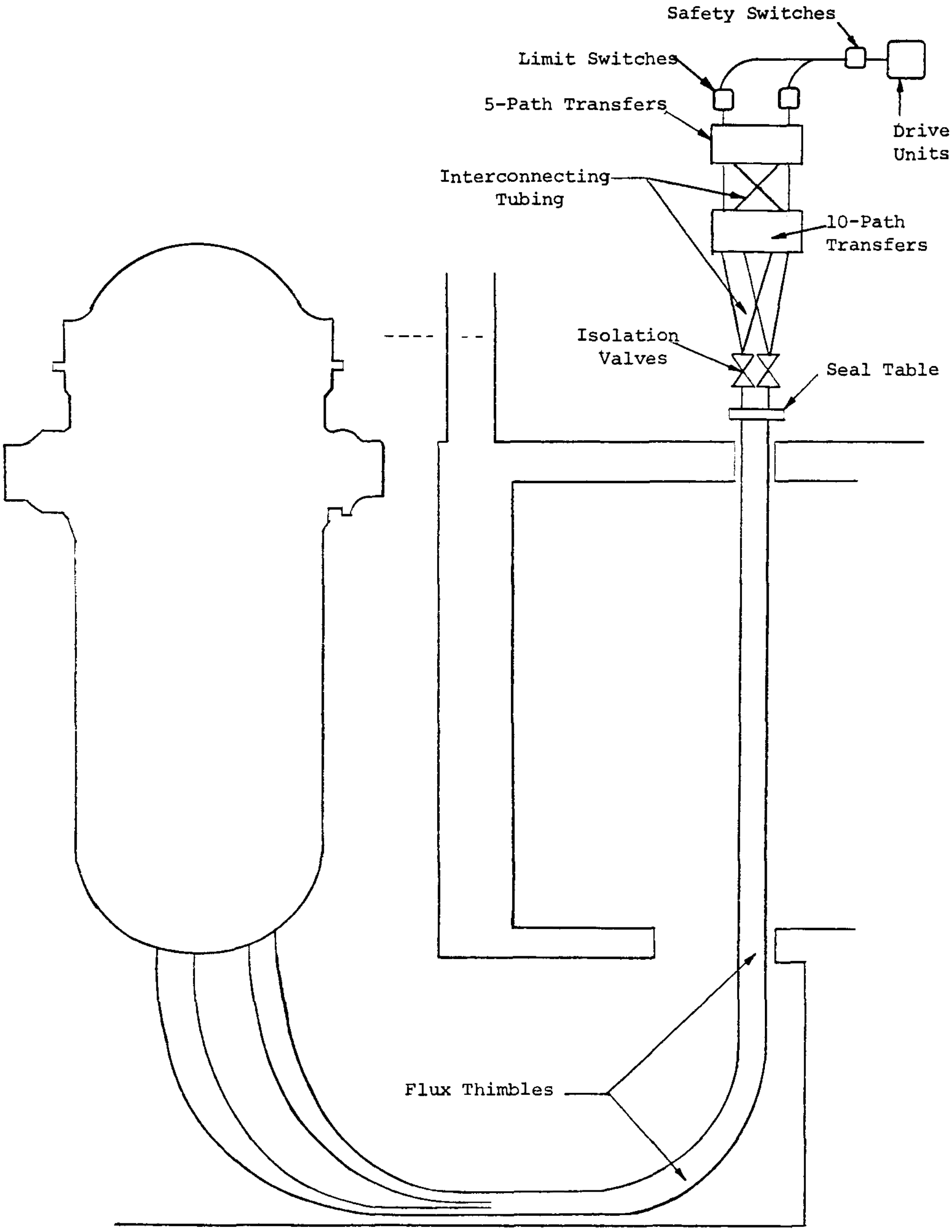


FIGURE 2.2.5

BASIC FLUX-MAPPING SYSTEM

Plotting is accomplished by driving the detectors to the top of their respective tracks, then slowly withdrawing them while plotting reaction rate versus position.

The control and readout system provides means to rapidly transverse the miniature neutron detectors to and from the reactor core at seventy-two feet per minute and to traverse the reactor core at twelve feet per minute. The control system consists of two sections: one physically mounted with the drive units, and the other contained in the control room. Limit switches in each tubing run provide signals to the path display to indicate the active detector path during the flux mapping operation. Each gear box drives an encoder for position indication. One five-path group path selector is provided for each drive unit to route the detector into one of the flux thimble groups or to storage. A ten-path rotary transfer assembly is used to route a detector into any one of up to ten thimbles. Manually operated isolation valves on each thimble allow free passage of the detector and drive cable when open. When closed, these valves prevent steam leakage from the core in case of a thimble rupture. Provision is made to separately route each detector into a common flux thimble to permit cross calibration of the detectors.

Chromel-alumel thermocouples are threaded into guide tubes that penetrate the reactor vessel head through seal assemblies, and terminate at the exit flow end of the fuel assemblies. The thermocouples are enclosed in stainless steel sheaths within the guide tubes to facilitate replacement when necessary. The thermocouples are located near the top of the fuel assembly and provide an indication of the moderator temperature at the exit of the fuel assembly.

2.2.1.2 Ex-Core Detectors

A. System Design:

The ex-core detector system primarily protects the reactor by monitoring neutron flux such that trips and alarms may be actuated based on pre-set conditions. The ex-core detectors are also used to indicate the status of various reactor parameters and provide a secondary control function. This detector system operates in 3 overlapping ranges: source, intermediate, and power. An increase in reactor power results in procedural increase in overpower protection level. Automatic re-set to more restrictive trip protection is provided when power is reduced. The ex-core detectors can monitor leakage neutron flux from shutdown to 120% power, and overpower excursions of up to 200% power can be recorded. For the purposes of this report, only the power range is of interest.

The ex-core detectors are calibrated daily against the thermodynamics of the plant, using the results of twice-daily calorimetrics across the secondary. In making in-core maps, an axial tilt is induced to calibrate the ex-core to the in-core in order to assure that the ex-cores are indicating the axial power split adequately.

The ex-core detectors, having been calibrated to the in-core detectors, are then used to monitor the various core behavior indices.

Alarm functions, but no control functions, are obtained from the source range channels. Both alarm and control functions are obtained from the intermediate range channels. Rod withdrawal may be blocked by either intermediate range or high flux level. Both alarm and control functions are obtained from the power ranges. Rod withdrawal is inhibited and an alarm is sounded by an over-power rod stop function from any of the four power range channels. Low flow trips and pump breaker trips are also actuated by the power range channels. A dropped control rod will be sensed by the power range channels. Power range channels also actuate deviation alarms, which alert the operator to a power unbalance between the power range channels, requiring corrective action. Finally, signals from the power range instruments are used in the process control system for the rod speed control function. The individual ion chamber signals are used for ΔT over-power/over-temperature compensation which initiates rod stops and turbine runbacks.

B. Physical Arrangement:

The ex-core detector system consists of eight independent channels: 2 source range, 2 intermediate range, and 4 power range. The relative positions with respect to the core of these ex-core detectors are shown in Figure 2.2.6. Testing and calibration of the detector channels are performed on-line.

The 8 channels of the ex-core detector system consit of 2 proportional counters, 2 compensated ion chambers, and 4 dual-section uncompensated ion chambers. Thus 8 detectors are placed in 6 radial locations around the core in the primary shield. This arrangement is illustrated in Figure 2.2.6. These detectors are placed on opposite core flats (Figure 2.2.6) at an axial point 25% from the core bottom. Intermediate range neutron sensors consist of compensated ion chambers placed in the same assemblies as the source range detectors, but positioned axially 50% from the core bottom. The four

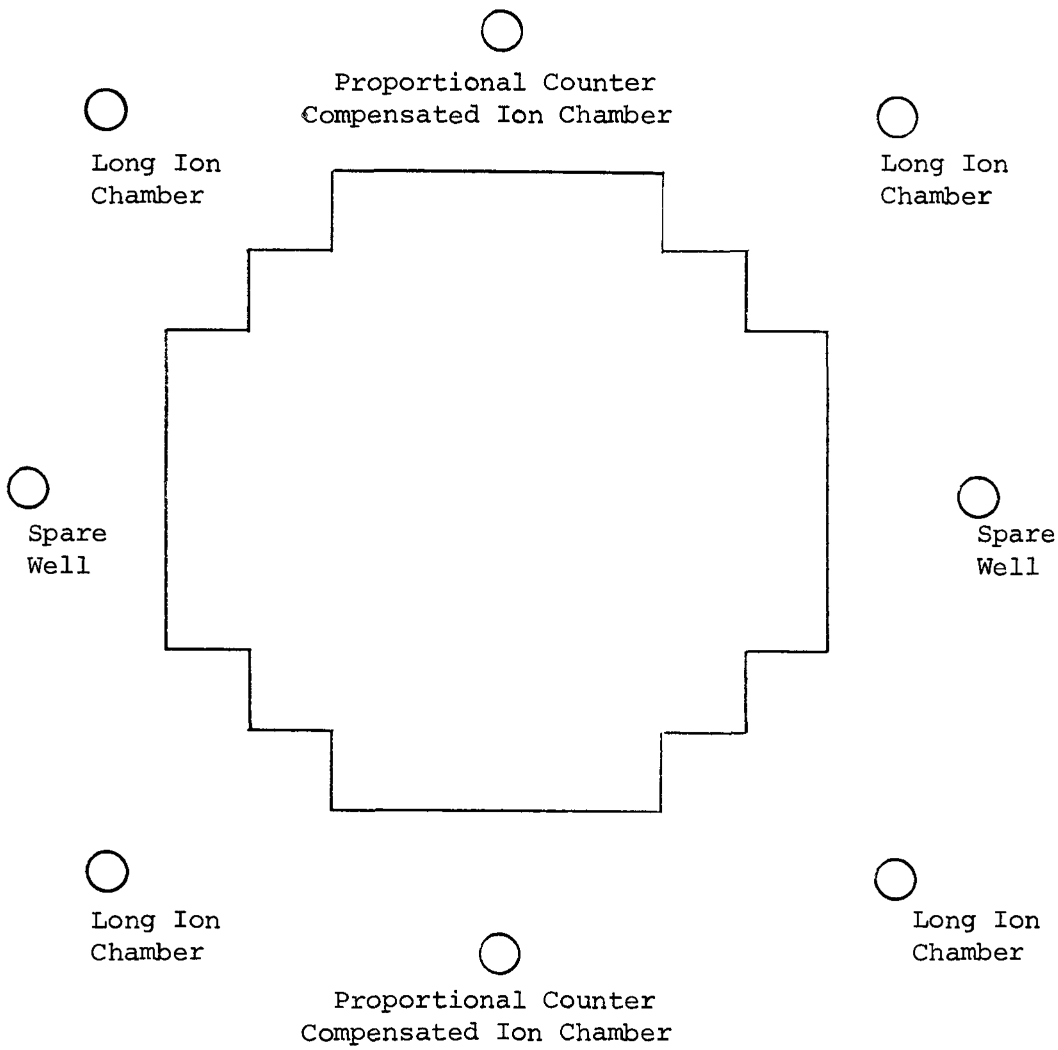


FIGURE 2.2.6

EX-CORE DETECTOR LOCATIONS

detector assemblies on the core diagonals, shown in Figure 2.2.6 contain the power range ion chambers. These assemblies are located within one foot of the reactor vessel to minimize flux distortion.

C. Type of Detector:

The four power range ex-core detectors each have a neutron sensitive length of 10 feet. Each detector assembly contains a pair of detectors (for monitoring the upper and lower halves of the core). By various combinations of the signals, the following information is derived:

- 4 indications of total power (upper plus lower)
- 4 indications of imbalance (upper minus lower)
- Deviation (maximum minus minimum of four)
- 1 upper azimuthal tilt (ratio of maximum to average - upper)
- 1 lower azimuthal tilt (ratio of maximum to average - lower)

All 8 signals from the four power range detector are continually recorded. Indications are provided for power and imbalance.

2.2.2 Sampling Rates

The movable in-core detectors must be used to determine the detailed power distribution on a monthly basis per the tech. specs. for most Westinghouse reactors. In practice, this in-core mapping is often done more frequently, especially for selected assemblies where the power peaks are occurring.

The ex-core instruments are calibrated daily against the thermodynamics of the plant to assure they are indicating actual core power level. The ex-core instruments are calibrated monthly when the in-core detectors map the core. They are calibrated to be sensitive to the power in the top half of the core relative to the power in the bottom half of the core. Using one minute averages of the instrument current, the axial offset is determined and monitored continuously. They are also calibrated to be sensitive to the relative power in the four different core quadrants. These quadrant power indicators are sampled continuously and checked for quadrant power tilts.

When the excore detectors indicate an axial offset outside specified bounds or a quadrant tilt outside specified bounds, tech. specs. require more detailed information so the core power distribution can be obtained and evaluated. For example, if the quadrant to average power tilt exceeds a certain value (2%), the hot channel factors shall be determined within 2 hours, or the reactor power must be reduced. (Presumably the in-core detectors are used to obtain the information via a core map and its conversion to a core power distribution).

2.2.3 Neutron Instrumentation Uncertainties

The experimental verification of power distribution analyses for Westinghouse reactors is discussed in Reference 6. The following summary of the report is taken from a W FSAR.[4]

"In a measurement of peak local power density, F_Q, with the movable detector system, the following uncertainties have to be considered:

(a) reproducibility of the measured signal

(b) errors in the calculated relationship between detector current and local flux

(c) errors in the calculated relationship between detector flux and peak rod power some distance from the measurement thimble."

"The appropriate allowance for (a) above has been quantified by repetitive measurements made with several inter-calibrated detectors by using the common thimble features of the incore detector system. This system allows more than one detector to access any thimble. Errors in category (b) above are quantified to the extent possible, by using the fluxes measured at one thimble location to predict fluxes at another location which is also measured. Local power distribution predictions are verified in critical experiments on arrays of rods with simulated guide thimbles, control rods, burnable poisons, etc. These critical experiments provide quantification of errors of types (b) and (c) above."

"Critical experiments have been performed at the Westinghouse Reactor Evaluation Center and measurement taken on two Westinghouse plants with incore systems of the same type as used in the plant described herein. The report concludes that the uncertainty associated with the peak nuclear heat flux factor, F_Q is 4.58% at the 95% confidence level with only 5% of the measurements greater than the inferred value. This is the equivalent of a 2σ limit on a normal distribution and is the uncertainty to be associated with a full core flux map with movable detectors reduced with a reasonable set of input data incorporating the influence of burnup on the radial power distribution."

"In comparing measured power distributions (or detector currents) against the calculations for the same situation it is not possible to subtract out the detectors reproducibility. Thus a comparison between measured and predicted power distributions has to include some measurement error. Since the first publication of the report, hundreds of maps have been taken on these and other reactors. The results confirm the adequacy of the 5% uncertainty allowance on F_Q according to Westinghouse." It shall be noted that in Ref. 5 it is stated that the number and location of thimbles was chosen to permit measureemnt of F_Q to within $\pm$ 10%.

2.3 Babcock & Wilcox Reactors

Figure 2.3.1 is a schematic showing the interrelation among the major components in the B&W monitoring system. These components are described in the following section. Portions of the descriptions have been taken from Reference 7 and other B&W descriptions of the system.

2.3.1 System Description

2.3.1.1 Fixed In-Core Detectors

The incore monitoring system provides neutron flux detectors to monitor core performance. Incore, self-powered neutron detectors measure the neutron flux in the core to provide a history of power distribution during power operation. Data obtained provides power distribution information and fuel burnup data to assist in fuel management. The plant computer provides normal system readout, and a backup readout system is provided for selected detectors.

The incore monitoring system consists of assemblies of self-powered neutron detectors located within the core. In this arrangement, an incore detector assembly consisting of seven local flux detectors, one background detector, and one thermocouple is installed in the instrumentation tube of each of the selected fuel assemblies. The local detectors are positioned at seven different axial elevations to provide the axial flux gradient. The outputs of the local flux detectors are referenced to the background detector output so that the differential signal is a true measure of neutron flux.

In a typical 1000 MWE core, the nuclear power in a fuel segment or node is monitored at 364 locations (52 assembl[illegible] have detectors at 7 axial locations) in the reactor core

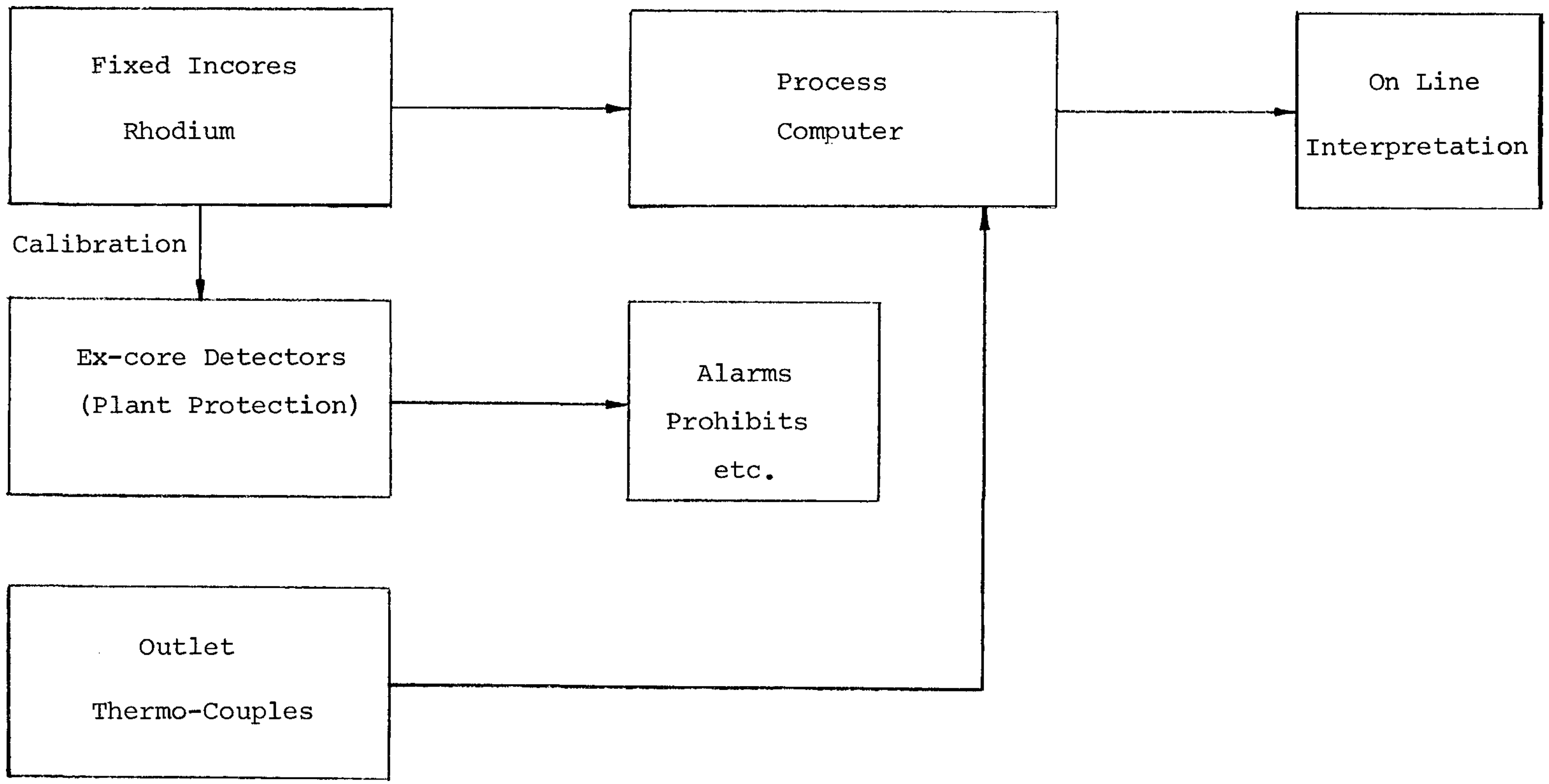


FIGURE 2.3.1

Block Diagram of Typical Babcock & Wilcox
Power Distribution Monitoring System

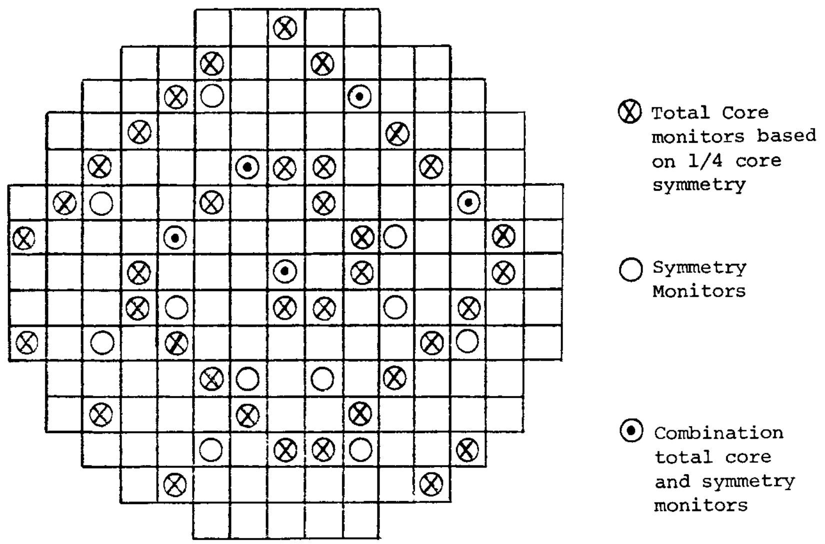
by the in-core monitoring systems' self-powered neutron detector assemblies. Thermocouples installed within these assemblies also monitor the exit temperature distribution at 52 locations in the core. The outputs of this system are connected to the plant computer and to in-core monitoring recorders which provide the operator with power distribution readouts.

Under normal operating conditions, the in-core detectors supply information to the operator in the control room.

Each individual detector measures the neutron flux in its vicinity and is used to determine the local power density. The individual power densities are then averaged and a peak-to-average power ratio calculated. This information can be used to indicate possible power oscillations.

B. Physical Arrangement:

The detectors are positioned at seven equally spaced axial locations in a fuel assembly (see Figure 2.3.2). In the radial or X-Y direction, the in-core instruments are located in approximately 25% of the assemblies. Figure 2.3.2 also shows these locations for a typical core. These in-core monitors are positioned so that there are three functional types:



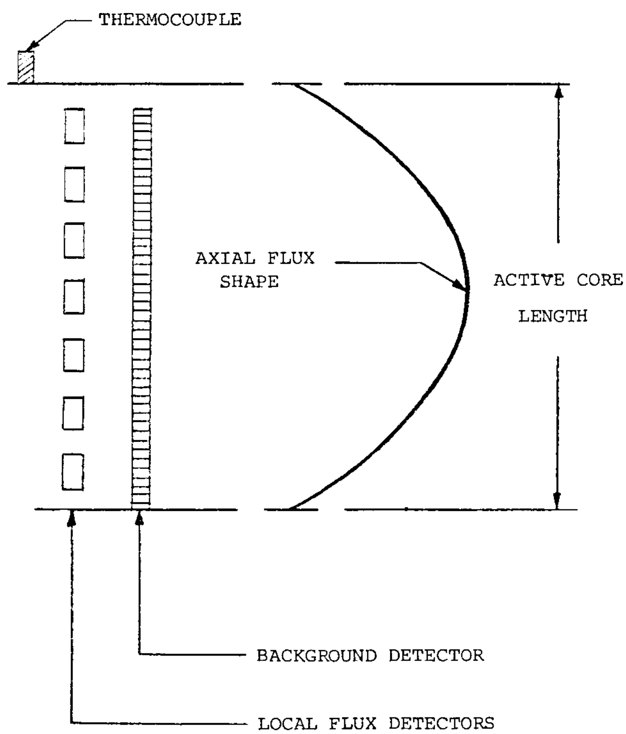


FIGURE 2.3.2

Radial Location and Axial Configuration of Typical

Babcock & Wilcox In-core Measurement System

1) Symmetry monitors - those which have a symmetrical monitor in each octant of the core.

2) Total core monitors - those which have been positioned to provide total 1/4 core coverage if quadrant symmetry is assumed.

3) Combination monitors - those few locations where the monitor is both a symmetry monitor and 1/4 core monitor.

C. Type of Detector:

Self-powered neutron detectors were developed by Atomic Energy of Canada Limited (AECL). The self-powered detector consists of a coaxial cable with a section of the central conductor replaced by a material that emits energetic electrons when exposed to thermal neutrons. A typical detector construction is shown in Figure 2.3.3. The energetic electrons penetrate the solid insulation and come to rest on the collector or its surroundings. The deficiency of electrons in the emitter results in a positive charge on the central conductor of the coaxial cable. The rate of this positive-charge production can be continuously measured by a current meter that connects the central conductor to the outer sheath of the coaxial cable through a resistor. The sensitivity, response time, and burnout rate of the detector depend on the choice of emitter material. For the case of rhodium as the emitter the neutron capture and decay is:

$$_{45}Rh^{103} + {_0}N^{1} \rightarrow {_{45}}\underset{4.4M}{Rh^{104}} \rightarrow {_{46}}Pd^{104} + \beta^{-}$$

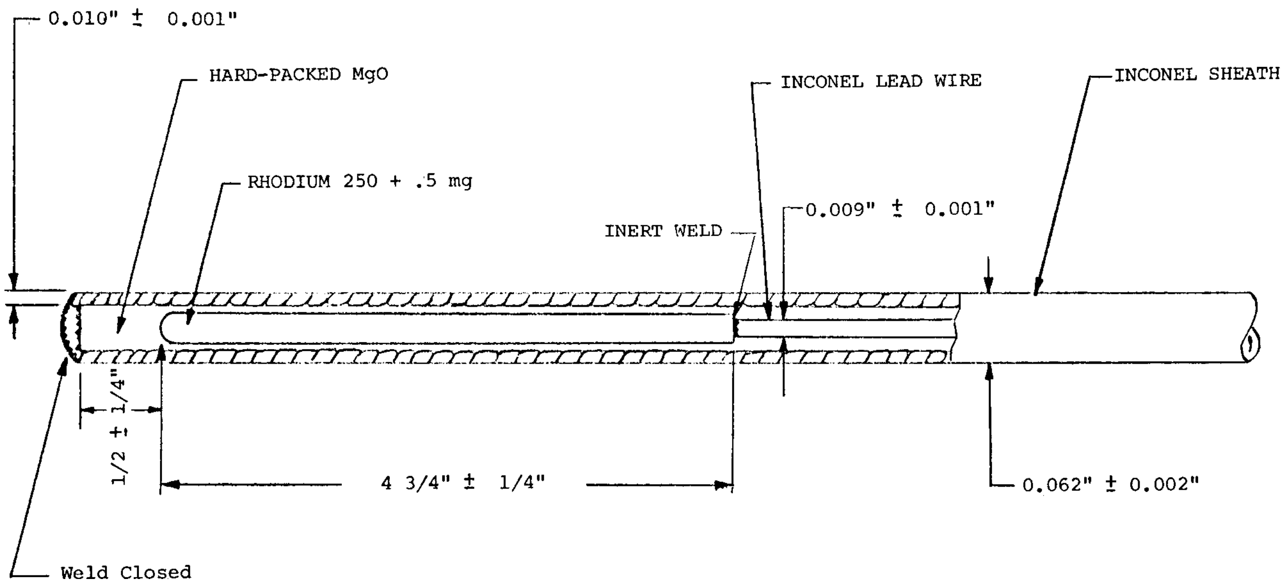


FIGURE 2.3.3

INCORE DETECTOR ASSEMBLY

Incore Self-Powered Neutron Detector

Since the sensitivity of these detectors is relatively low (compared to fission detectors), care must be exercised in the design of the detector and the signal-transmitting and -conditioning system to prevent unacceptable background levels.

The background detector utilizes an Inconel detector core without rhodium. The outputs of the rhodium detector are referenced to the background detector output so that the differential signal is a true measure of the neutron reaction rate.

Readout for the incore detectors is performed by the plant computer. Multipoint recorder readouts of selected detectors are provided independent of the computer.

When the reactor is depressurized, the incore detector assemblies can be inserted or withdrawn through guide tubes which originate at a shielded area in the Reactor Building. These guide tubes enter the bottom head of the reactor vessel where internal guides extend up to the instrumentation tubes of 52 selected fuel assemblies. The instrumentation tube serves as the guide for the incore detector assembly. During refueling operations, the incore detector assemblies are withdrawn approximately 13 feet to allow free transfer of the fuel assemblies. After the fuel assemblies are placed in their new locations, the incore detector assemblies are returned to their fully inserted positions.

The nature of the detectors permits the manufacture of nearly identical detectors which produces a high relative accuracy between individual detectors. The detector signals are compensated continuously for burnup of the neutron sensitive material by the plant computer.

According to B&W (Ref. 7), calibration of detectord is not required. The incore self-powered detectors are controlled to precise levels of initial sensitivity by quality control during the manufacturing stage. The sensitivity of the detector changes over its lifetime due to such factors as detector burnup, control rod positions, fuel burnup, etc. The results of experimental programs to determine the magnitude of these factors have been incorporated into calculations and is used to correct the output of the incore detectors for these factors. Operation of detectors in both power and test reactors has demonstrated that this eompensation program, when coupled with the initial sensitivity, provides detector readout accuracies sufficient to eliminate the need for a calibration system according to the B&W position. However, in actual operating reactors there are significant variations in detector output from symmetrical locations.

The temperature detector is a special-grade, type K, grounded-junction thermocouple of chromel-alumel.

2.3.1.2 <u>Ex-Core Detectors</u>

A. System Design:

The ex-core power distribution monitoring system utilizes 4 detector assemblies each containing two uncompensated ion chambers. These ion chambers are positioned so as to represent the top and bottom half of the core.

The summation of the top and bottom signals represents the total reactor power (0 - 125% full power) while the difference represents power imbalance (±62.5%).

The signals are averaged in pairs. The higher average is used to represent core power. These detectors are also used to monitor quadrant symmetry, again based on periodic calibration by the in-core instruments.

It is important to note that the system output only indicates relative core power and must be calibrated against reactor calimetrics and in-core detectors. This assures that total core power level is being monitored as well as gross spatial characteristics.

B. Physical Arrangement:

Each assembly consists of two 72 inch uncompensated ion chambers with a single high voltage connection and two signal connections. Signals proportional to the sum and the difference of the signals are derived.

The physical location of the detectors are shown on Figure 2.3.4. These detectors are located external to the core and monitor the core by monitoring the flux changes at the detector location. Detectors NI-5 and NI-6 are averaged as are NI-7 and NI-8. The averages are then auctioneered such that the higher average represents core power.

Table 2.3.1 provides the characteristics of the ex-core detectors.

2.3.2 <u>Sampling Rates</u>

In a typical B&W plant, the determination of the in-core power distribution from the in-core instruments is required as follows:

- At low power during initial reactor startup to check that power distribution is consistent with calculations
- Subsequent checks during operation each 4,000 MWD/MTU average burnup to insure that power distribution is consistent with calculations
- Indication of power distribution in the event that abnormal situations occur during reactor operation; e.g. an inoperable rod

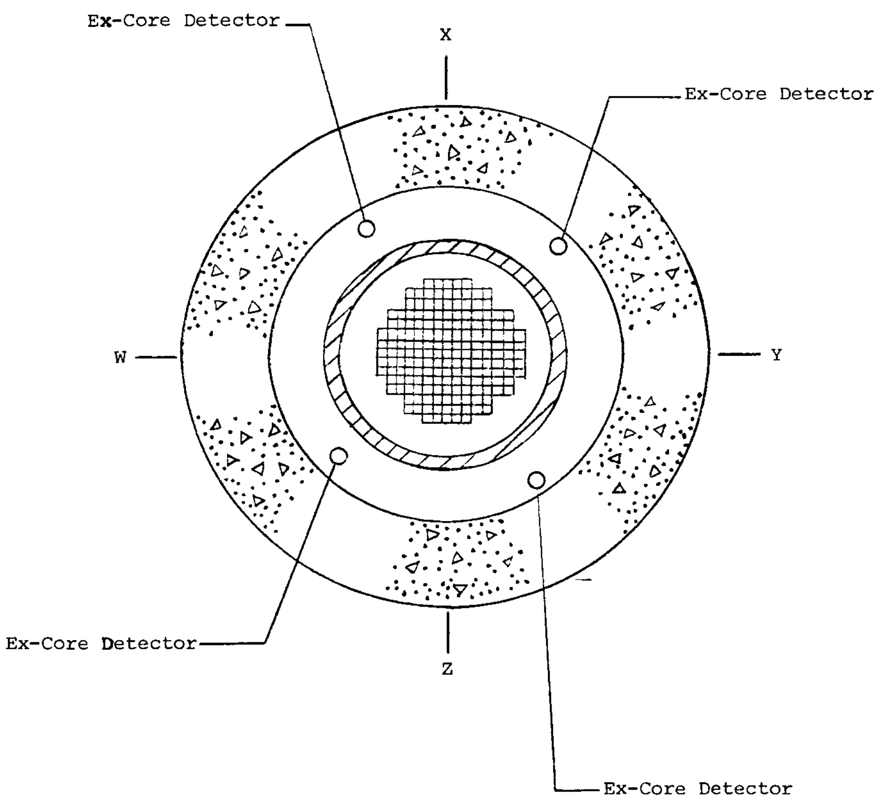


FIGURE 2.3.4

LOCATION OF EX-CORE DETECTORS

TABLE 2.3.1

CHARACTERISTICS OF OUT-OF-CORE NEUTRON DETECTOR ASSEMBLIES

Characteristic	Power
Tube type	UCIC
Assembly No.	WL-23636B
<u>Sensitivity</u>:	
Thermal neutron flux	1.62×10^{-3} A/nv
Gamma flux	1.14×10^{-10} A/R/h
<u>Maximum Ratings</u>:	
External pressure	150 psig
Temperature	212 F
Thermal neutron flux:	
Operating	2.5×10^{10} nv
Non-operating	2.5×10^{11} nv
Gamma flux	5×10^{5} R/h
<u>Integrated Exposure Before 10% Reduction in Sensitivity:</u>	
Neutron	10^{19} nvt
Gamma	3×10^{9} R

- Re-calibration of ex-core detectors on a monthly basis
- A comparison check between in-core and ex-core results when one of the four out-of-core detectors indicate an asymmetry

The ex-core detectors are used to determine axial imbalance and quadrant tilts on a short term periodic basis. In a typical B&W plant the quadrant tilt must be monitored every two hours when the plant is above 15% power; and the axial imbalance must be monitored every two hours above 40% power.

2.3.3 Neutron Instrumentation Uncertainties

The accuracy of the detector output relative to the other detectors in the core depends upon strict manufacturing tolerances associated with the mass of the detector. The original objective of B&W was that these tolerances would contribute an error of not more than 1% in the detector output signal. Upon initial installation, the detector has the capability to measure the relative flux with an accuracy of 5% when used in conjunction with a background detector. The sensitivity of the detector will decrease with exposure to neutron flux due to depletion of rhodium. However, by use of integrated current inventories, it is felt that the additional inaccuracies are no more than 1% per year for the average flux condition.

However, it is the author's experience that the variation in detector output at symmetrical locations can be significantly larger than the expected 5% to 6%.

2.4 Combustion Engineering Reactors

Figure 2.4.1 is a schematic showing the interrelation and intereaction among the major components in the C.E. monitoring system. These components are described in the following section. Major portions of this section were taken from references 8, 9, 10, 11, and 12.

2.4.1 System Description

2.4.1.1 Fixed and Movable In-Core Detectors

A. System Design:

The in-core nuclear instrumentation which is being installed in the present generation of C.E. power reactors is intended to provide supplementary information about the gross power distribution within the reactor core but it plays no role in the protection of the reactor plant since the design limits on control rod insertion, maneuvering rates, and other operational characteristics are determined during the design of the core, with a more-than-adequate margin. In addition, the in-core instrumentation is not felt to be necessary to the economic operation of the reactor. Nevertheless, it is desirable in a large reactor to have more detailed information about the irradiation of the fuel assemblies than can be obtained from the out-of-core instrumentation in order to provide a means of verifying the validity of the basic design calculations.

The above is from Ref. 9; however, Ref. 11 has the following, which is a more positive utilization.

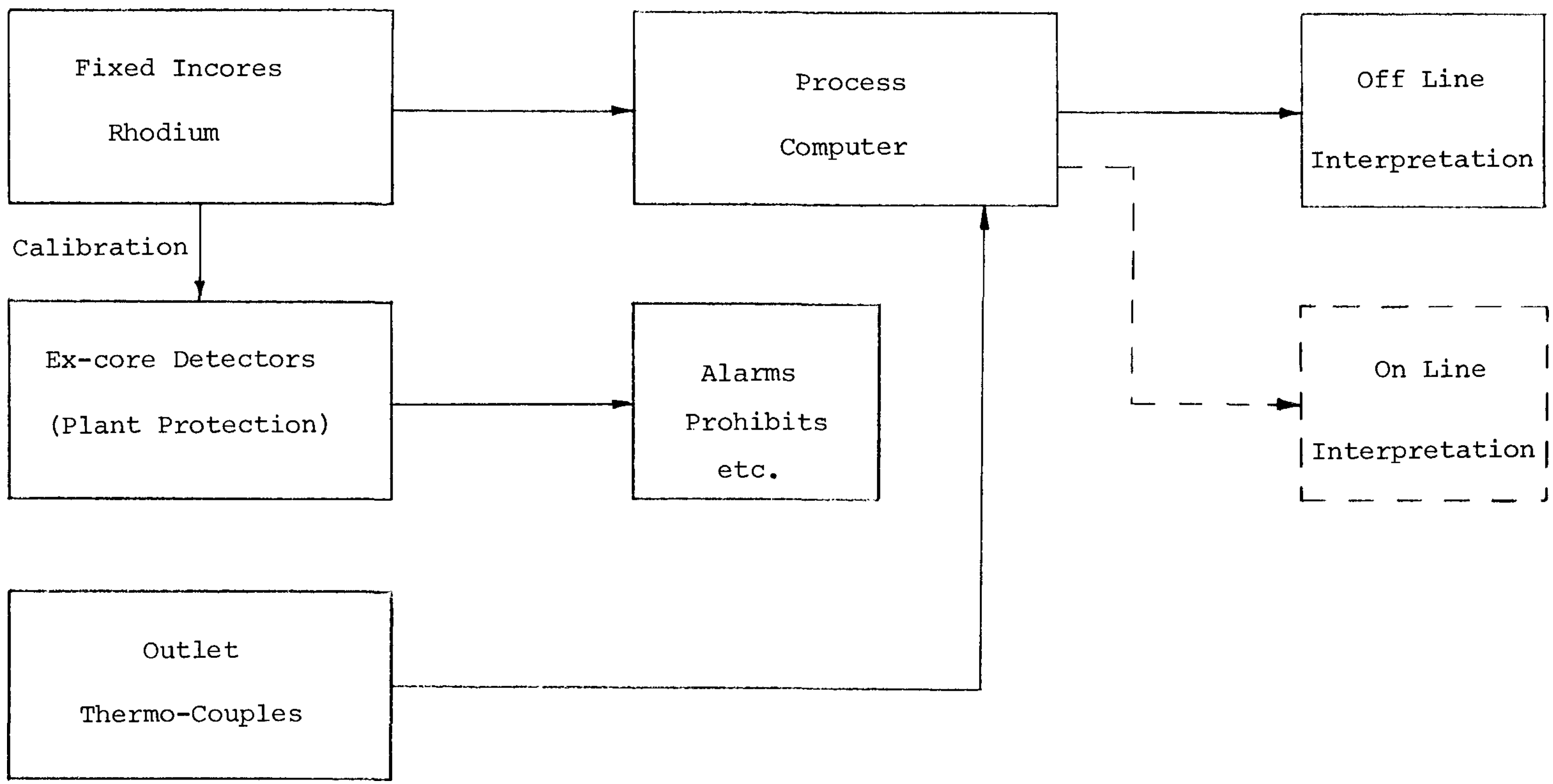


FIGURE 2.4.1

Block Diagram of Typical Combustion Engineering
Power Distribution Monitoring System

The use of ex-core detector measurement, though conservative, provides an acceptable basis for protection systems to monitor safety limits and provide trip signals to insure that the limits are not exceeded. However, more precise estimates of core power distribution are desirable for systems to monitor operating limits, which are generally of lower magnitude.

More precise measurements of core power distribution can be obtained directly from in-core detector signals. By utilizing a set of fixed in-core detectors, a "snap-shot" of the core power distribution can be obtained. Instrument signals are processed to obtain the peak pin power and hot channel heat flux in each assembly.

The instruments presently provided in Combustion reactors are Rh self-powered detectors which measure neither the flux nor the fission power, but the Rh activation. Consequently, it is necessary to apply theoretical corrections in order to convert the in-core instrument signals to more meaningful information. An analyses method is needed to perform this information conversion operation in a systematic way and to present the results in a form which has direct significance to the reactor engineer. In essence, the analyses converts instrument signals into the integrated power in each fuel assembly at any point in core life and the maximum linear heat rate for any fuel rod in a given fuel assembly. In addition, the accumulated exposure is provided on both an assembly-wise and batchwise basis.

The thermal in-core monitoring system comprises four types of detector assemblies where:

- Type 1 contains 4 Rhodium self-powered neutron detectors, 1 Vanadium self-powered neutron detector, and 1 Chromel-Alumel (Cr-Al) thermocouple.

- Type 2 is identical to Type 1, except that the Vanadium detector is replaced by a background detector.
- Type 3 contains 4 Rhodium detectors, 1 Vanadium detector, 1 Cr-Al thermocouple, and 1 calibration tube to permit the use of movable in-core detector in this core location.
- Type 4 is identical to Type 3, except that the Vanadium detector is replaced by a background detector.

These are shown as a composite in Figure 2.4.2 for a reactor that recently became operational. The Vanadium detectors are used to obtain a measure of the integral of the axial flux in the instrument tube for those assemblies with the Vanadium.

B. Physical Arrangement:

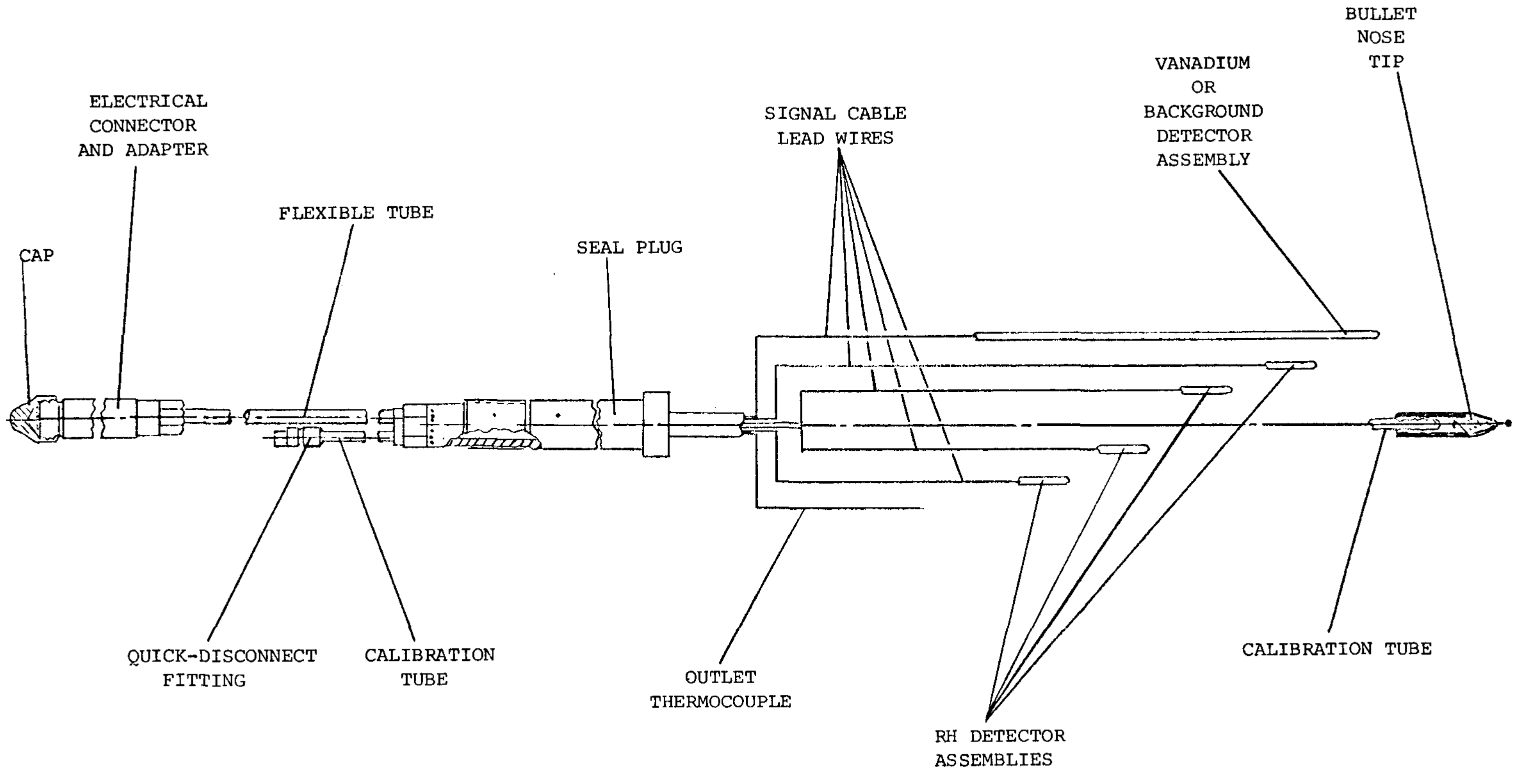
The fixed detectors are positioned at four axial positions with their centers spaced at 20, 40, 60 and 80% of the core height. The movable in-core detector provides signals for all axial locations in the fuel assemblies that accommodate such detectors. Figure 2.4.4 shows the axial location of the fixed detectors and depicts the movable in-core detector (those locations with a calibration tube) for the Types 3 and 4 detector assemblies.

In the radial or X-Y direction, the fixed in-core instruments are located in approximately 20% of the assemblies. Figure 2.4.3 shows these locations for a typical C.E. core. The four types of detectors that are described in Section A above are identified by different symbols.

The monitors are arranged such that each quadrant has six monitor locations in common with all other quadrants. When these are combined with the remaining detectors in one octant, they provide the capability of mapping ∿55% of the fuel assemblies (assuming octant symmetry), or

FIGURE 2.4.2

MODIFIED IN-CORE DETECTOR ASSEMBLY



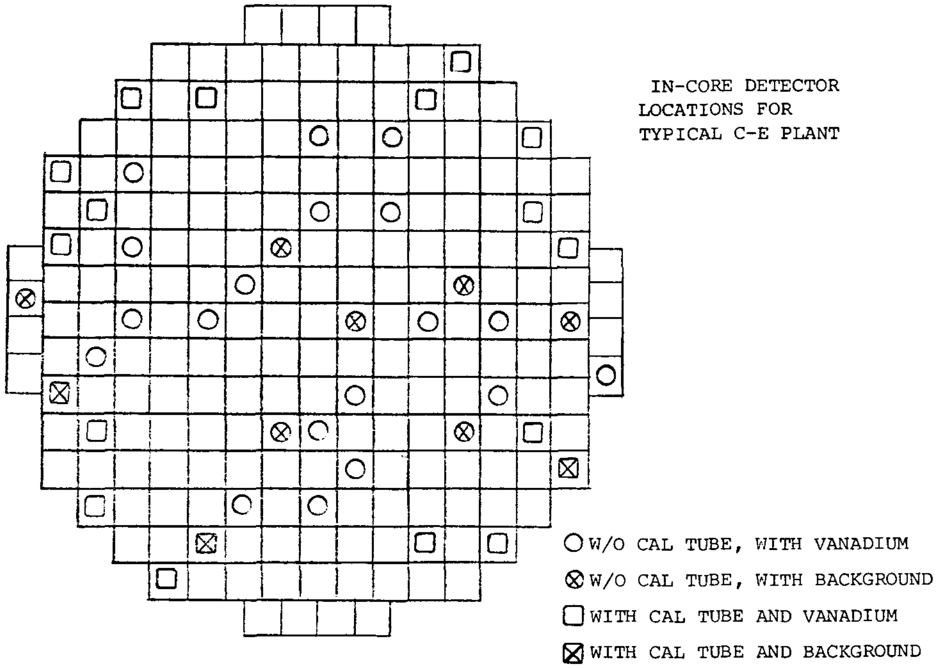


FIGURE 2.4.3

IN-CORE DETECTOR
LOCATIONS FOR TYPICAL
C-E PLANT

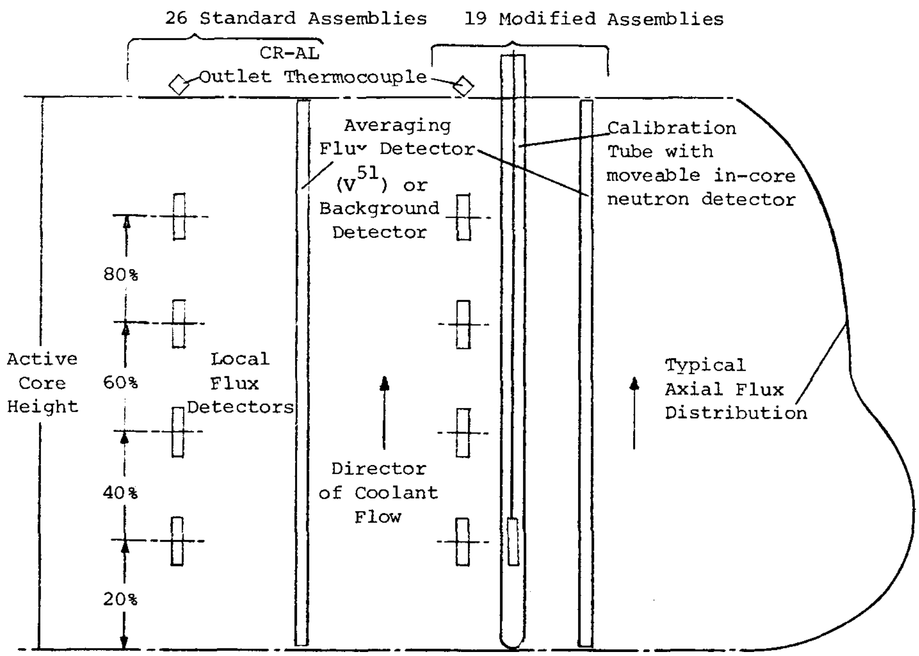


FIGURE 2.4.4

AXIAL CONFIGURATION OF C-E IN-CORE NEUTRON DETECTOR ASSEMBLY each type of assembly not occupied by a Control Element Assembly.

The modified assemblies near the periphery contain a calibration tube of full core height. Accordingly, the movable in-core detector can map any axial location in these assemblies. Only these peripheral assemblies contain a calibration tube due to the bend radii restrictions within a reactor vessel. The thermocouples are located at the top of the fuel assemblies, so as to measure the outlet coolant temperature of the instrumented fuel assemblies.

The vanadium and background detectors are of full core height. Vanadium detectors are made long to get a usable output signal, since Vanadium has a lower sensitivity than Rhodium. Being full length, the vanadium detector output is proportional to the integrated fuel assembly power. The background detectors will indicate the background signal component that will be seen by the Rhodium detector located at the bottom of the assembly.

C. Type of Detector:

The principle of operation of self-powered neutron detectors involves conversion of the incident neutron radiation on the detector emitter material to energetic electrons which penetrate the solid insulation and come to rest on the collector or its surroundings. The deficiency of electrons in the emitter results in a positive charge on the center conductor of the coaxial cable attached to the emitter. The rate of positive charge production produces a current which is directly proportional to the rate at which radiation is being absorbed by the emitter.

As described in Section 2.3 of the B&W system the primary mechanism by which the incident neutron radiation in the rhodium and vanadium detector is converted to energetic electrons is through neutron capture in the emitter producing a capture which decays through beta emission. Some of the beta particles are energetic enough to escape from the emitter resulting in a positive charge on the emitter.

The rhodium and vanadium emitters have reasonable large neutron capture cross sections, and their capture products are beta emitting isotopes with short half-lives. Rhodium-103, for example, has a 150 barn 2200 m/sec cross section and its capture product (rhodium-104) emits beta particles possessing an end-point energy of 2.4 Mev with a 42 second half-life.* Vanadium-51 has only a 5 barn 2200 m/sec absorption cross section and its capture product (vanadium-52) emits beta particles having a 2.5 Mev end point energy . Since the cross section of the vanadium detectors is much less than rhodium, its sensitivity per unit length (and therefore its burnout rate) is much less than rhodium. However, the vanadium detectors are full core length while the rhodium detectors are only 40cm long causing the total sensitivity (to 2200 m/sec neutrons) to be slightly greater than that of rhodium.

A sketch of the instrument assembly is shown in Figure 2.4.2. Note that the "fifth" neutron detector in the assembly can be either a full core length vanadium detector, or a full length section of cable for background detection.

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* An isometric state, RH^{104m}, is also produced approximately 7.3% of the time with a 4.41 min. half-life.

Table 2.4.1 below gives a comparison of various properties of the rhodium and vanadium detectors for a PWR per C.E. description.

As shown in Figure 2.4.5, a self-powered detector consists of an emitter (which emits electrons upon neutron interaction), a collector, and insulation.

The rhodium detectors each are 40cm long and approximately 0.064" OD, with Al_2O_3 insulation, unlike B&W which use M_gO_2. The detectors are of integral construction (the detector sheath and signal cable sheath are made of one piece of tubing with no joint between the detector and signal cable).

The Vanadium detectors each have a 135" long emitter made of V^{51}, an Inconel sheath of approximately 0.083" OD, and Al_2O_3 insulation.

The background detectors are simply lengths of signal cable, with no emitter. These are used to determine how much signal is produced in the Rhodium detectors from neutron and gamma interactions with the signal cables.

The movable in-core detectors have been Rhodium self-powered, which is similar to the fixed Rhodium detectors and miniature fission chambers similar to those described in Section 2.2.1.

These detector assemblies are inserted through the instrumentation nozzles. They then go into guide tubes, and finally into zircaloy thimbles within the fuel assemblies.

TABLE 2.4.1

COMPARISON OF PROPERTIES OF RHODIUM DETECTORS WITH THOSE OF VANADIUM DETECTORS

	Property	$Rhodium^{103}$ Detector	$Vanadium^{51}$ Detector
1.	Length	40 cm	336 cm
2.	Diameter	18 mils	37 mils
3.	Sensitivity	0.915×10^{-21} amp/nv-cm	1.75×10^{-22} amp/nv-cm
4.	σ_1 (PWR Spectrum)	9 b	.06 b
5.	σ_2 (PWR Spectrum)	75 b	2.7 b
6.	Depletion Rate in 10^{13} n/cm^2-sec (Maxwellian)	0.23% per month	0.013% per month
7.	Depletion Rate in PWR Spectrum $(\sigma_1\phi_1+\sigma_2\phi_2)$ x 1 month	1.2% per month	0.03% per month

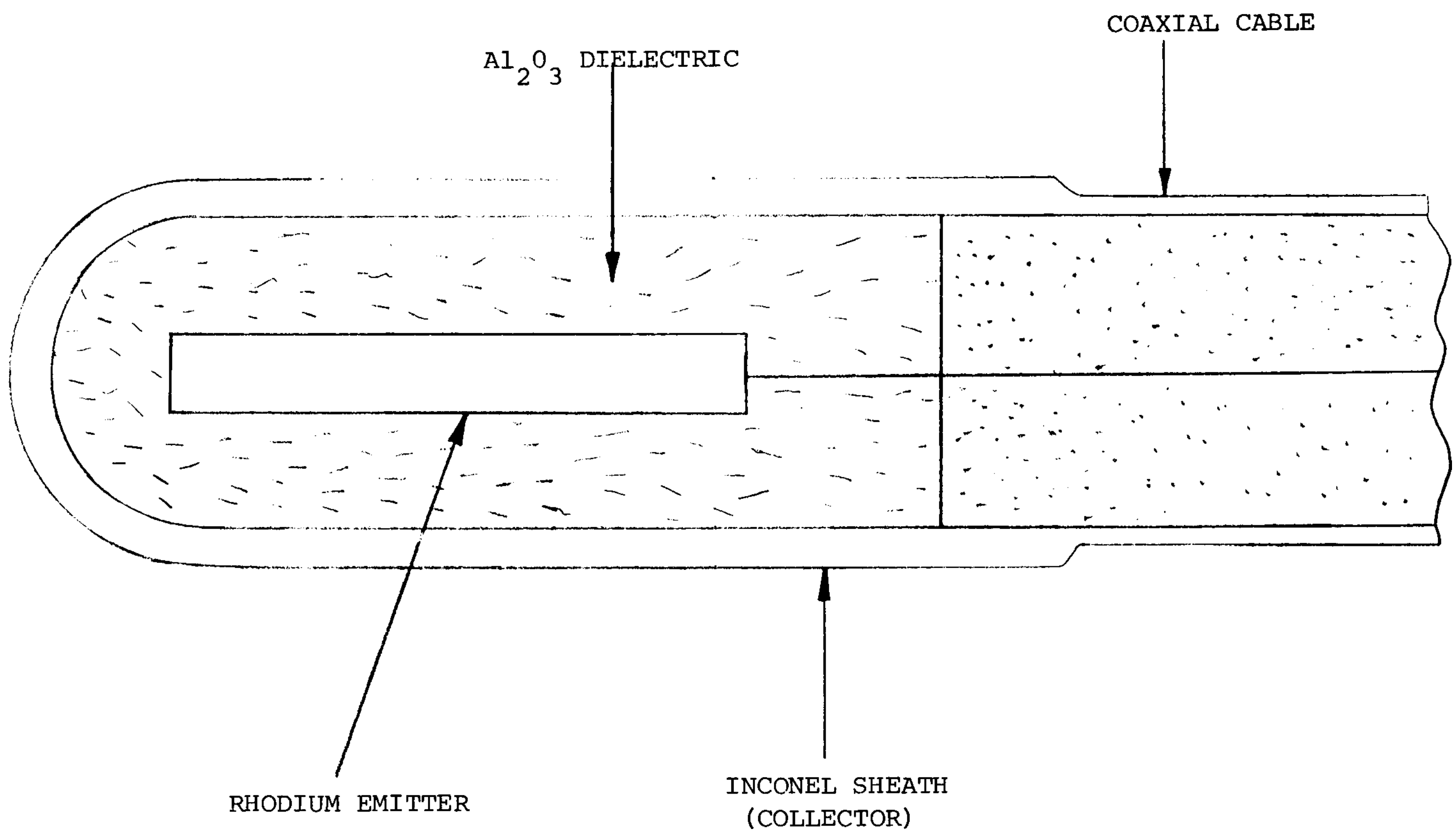


FIGURE 2.4.5

TYPICAL RHODIUM DETECTOR

The details within the vessel are such that the detectors have to be bent to reach the required thimbles, and that the degree of bending required varies with different assemblies.

The assemblies containing calibration tubes can only go into locations which have bend radii of equal or greater than 24 inches.

2.4.1.2 Ex-Core Detectors

A. System Design:

C.E., in Ref. 11, described the monitoring philosophy of ex-core detectors as follows.

On-line monitoring of core power distribution in both monitoring systems and protection systems has in the past been provided through the use of two level ex-core detectors. Design calculations were used to establish a correlation between the upper limit on expected peaking and the lower to upper axial power split, or axial shape index (Table 1.1). The peaking predicted by this correlation must be sufficiently conservative to accommodate the uncertainties resulting from the geometrical "shape annealing" associated with the placement of detectors away from the core, from the shadowing effects associated with measurements based on peripheral fuel bundles rather than the entire core, and from the low resolution associated with the use of only two levels of flux information. The use of two level ex-core detectors and the associated correlations to establish peaking lead to the familiar "tent curves" which have been included as technical specification monitoring band limits.

The use of three levels of ex-core detector information helps alleviate the uncertainty due to resolution by allowing one to construct an approximate axial power distribution rather than depending on a correlation between peaking and axial shape index. The use of ex-core detector measurements, though conservative, provides an acceptable basis for protection systems to monitor safety limits and provide trip signals to ensure that these limits are not exceeded. However, more precise estimates of core power distribution are desirable for systems to monitor the operating limits, which are generally of lower magnitude.

B. Physical Location:

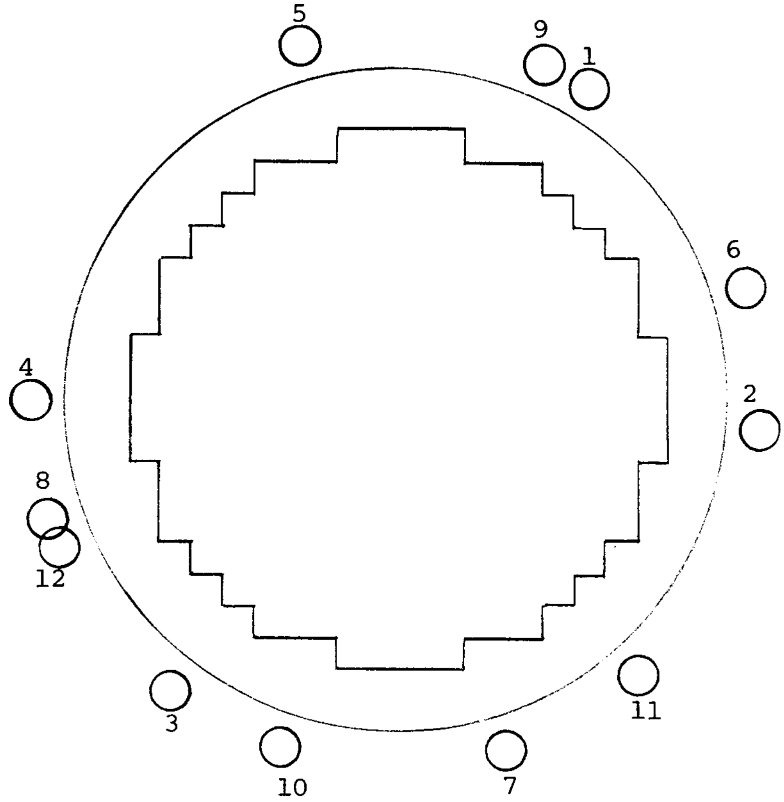
The locations of the out of core detectors relative to the core and each other, are shown in Figure 2.4.6. This configuration was obtained from a C.E. plant that recently started and all the ex-core detectors are shown in the figure.

2.4.2 Sampling Rates

According to the tech. specs. of one C.E. reactor, "The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the ex-core detector monitoring system or with the in-core detector monitoring system."

In order to demonstrate this, it is required to perform a core-power distribution map at least once per 31 days; and to verify the ex-core detector outputs via this map.

The total radial peaking factor (F_r^T) and the azimuthal power tilt (T_q) are monitored continuously. If T_q is outside the predetermined limit then the tilt must be corrected within the next 2 hours or determined within the next 2 hours that the total radial peaking factor (F_r^T) is within limits. Presumab this would be done by a core map. In addition, the T_q must be



	CHANNEL	ANGLE	RADIUS
1	Wide Range MA	29° - 30'	9' 6 1/2"
2	Wide Range MB	95°	9' 6 1/2"
3	Wide Range MC	220°	9' 6 1/2"
4	Wide Range MD	270°	9' 6 1/2"
5	Safety MA	345°	8' 10 3/32"
6	Safety MB	75°	8' 10 3/32"
7	Safety MC	165°	8' 10 3/32"
8	Safety MD	255°	8' 10 3/32"
9	Control RRS #1	22°	8' 10 15/16"
10	Control RRS #2	202°	8' 10 15/16"
11	Spare	140°	9' 6 1/2"
12	Spare	249°	9' 5"

FIGURE 2.4.6

EX-CORE DETECTOR LOCATION

determined using in-core detectors at least once per 12 hours when one ex-core channel is inoperable and the thermal power is above 75% of rated.

2.4.3 Neutron Instrumentation Uncertainty

The accuracy of the in-core detectors have been assessed in at least one C.E. core by intercalibration of the in-core detectors. The procedure and results were presented in Ref. 10.

Detector intercalibration was carried out by the vendor. All detectors were calibrated relative to standard detectors. The relative calibrations were checked by repeating them with the detectors in different positions, holders, reactors and after 2-8 months of elapsed time. The resulting standard deviation in the relative intercalibration factors is 0.25%.

Absolute calibration of the detectors was carried out using cobalt wires attached to the outside of selected detectors for exposure in standard conditions. The wires were absolutely counted and their activations interpreted to give the absolute flux at the detector location. The uncertainty in the measured absolute flux was ±4-6% which, in turn, is reflected as a similar uncertainty in the absolute sensitivity of the detectors. Absolute calibration of the detectors, although useful, is by no means as important as their relative intercalibration since the reactor power is basically determined by calorimetric measurements and calculations.

An additional measure of the reproducibility of the detector signals under operating conditions was obtained by accumulating a large number of readings during power operation at Palisades.

During several periods of steady state operation, repeated measurements were obtained from all 180 rhodium in-core detectors. After known bad detectors and those affected by slight control rod movements had been eliminated, about 4800 duplicate measurements of detector signals were compared. By comparing sequential pairs of measurements, the standard deviation between pairs of readings was found to be 0.35%, averaged over the total data set.

In at least one reactor the author is aware of, however, there appears to be as much as 10% variation between the data from a movable fission detector and the data from the fixed in-core detectors.

3.0 ACQUISITION AND DATA QUALITY REQUIREMENTS

Section 2.0 described the in-core detector system of each LWR vendor from the viewpoint of design objectives, arrangement of detectors, type of detector and sampling rates. This section addresses the acquisition of the data or data quality requirements insofar as it is effected or influenced by such items as: the number and placement of detectors and the symmetry considerations involved; detector failure and impact on operation; flux gradients; detector depletion; time response of detectors; etc.

There is some repetition of descriptions among Sections 2, 3 and 4, (e.g., detector arrangement), however, the repetition helps to clarify individual descriptions in each section.

3.1 General Electric

3.1.1 Detectors

As described in Section 2, the in-core monitoring system consists of fixed in-core detectors and traversing in-core detectors. The instrument tube (which accommodates both fixed and movable detectors) is located in a water gap at the corner of four adjacent fuel assemblies (Fig. 2.1.4). The fixed detectors are positioned at four equally spaced locations axially. A typical X-Y arrangement of detectors is shown in Figure 2.1.2. This X-Y arragnement of detectors is such that when all the detector locations are rotated into one quadrant, all interior fuel assemblies in that quadrant are adjacent to a detector (the exterior or peripheral assemblies are not monitored). If there is fuel assembly and control rod symmetry in the core, then there is 1/4 core symmetry and $\sim$90% of the fuel assemblies (all those in high power locations) are "monitored." The locations that do not have an instrument tube and use the actual instrument signals from some other symmetrical locations in the core are called "pseudo" locations. In some BWR cores (such as Browns Ferry and Peach Bottom 1st cycles), the fuel assembly types, as characterized by fuel enrichment, burnable poison locations, etc., are not arranged such that quadrant symmetry is preserved. This means that when an LPRM string is rotated to another quadrant, the assemblies adjacent to the pseudo location are different than those adjacent to the actual location. This can further mean that the true axial shape in the pseudo location is different than that in the actual location and hence, the LPRM or TIP value would be different. In this instance, an analytic procedure is used to convert the actual signals in symmetric locations (either TIP or LPRM) to pseudo signals that will more nearly reflect the assemblies adjacent to the pseudo location. If there is an asymmetric condition due to rod configurations, the rotation is not done and algorithms are used to infer the pseudo signals using the actual locations as a base. It follows that the uncertainty in the "pseudo" location signals increases the further the core is from symmetry.

The power distribution in a BWR is determined from the fixed in-core detectors (LPRMs). These detectors provide a current which is proportional to the U-235 fission rate at the detector location. The LPRMs are intercalibrated by use of the TIP system. There are 3, 4 or 5 TIPS, depending on the core size, which are capable of traversing each LPRM string location; each TIP covers a fraction of the core. One instrument tube location, near the center of the core, can be accessed by each TIP so that each TIP can be normalized to a common value.

With the TIPS thus normalized, each instrument location is traversed. The individual LPRMs are made to give the same signals as the particular TIP did when it passed the LPRM location. In this manner, all the LPRMs are intercalibrated relative to one another.

Through a complicated calculational procedure, the LPRMs are related to absolute power (watts/cm) and reflect the average power in the 4 adjacent pins, or in the later versions the average power in the four nodes (axially segments of an assembly) adjacent to the instrument.

This procedure is discussed in some detail in Section 4.0, and is briefly summarized here for continuity. The LPRMs are intercalibrated by the TIPS and the TIP values (48 axial values for each instrument tube location) are stored. These axial TIP values are used in converting the 4 axial LPRM values into 48 axial values (the stored TIP values are used until a new map is done or an individual TIP is run into one channel to establish a new axial shape). The values at pseudo locations are established from those at actual instrument tube locations. These relative instrument readings (including the axial and X-Y inferred values) are related to the relative power in the adjacent assemblies by a series of factors that account for voids, exposure, type of fuel assembly, control rod proximity, etc. The total core power is obtained from a heat balance and apportioned to the monitored assemblies (including pseudo monitored assemblies) and un-monitored assemblies; thus allowing the relative power deduced from the instrument signals to be assigned absolute values.

The accuracy of the power distribution that results is dependent on a number of factors. For example:

- The axial shapes are very dependent upon rod positions, power level and recirculation flow.

- The longer the time since the last TIP core map, the greater the chance that the axial shapes have changed and the more uncertain the axial interpolation between LPRMs becomes.

- The LPRMs reflect the fission rate of U-235 at the detector location and converting this reaction to the fission rate (i.e., power) in adjacent assemblies is not straightforward and simple. Since the assembly fission rate is due to Pu and U-238 as well as U-235, these factors are a function of steam void and isotopic content (exposure).

- The environment surrounding the assemblies also affect conversion; the presence of control rods and the reactivity in the assemblies adjacent to those of interest can affect the conversion.

3.1.2 Reliability and Impact of Failure

The LPRMs are linked to the plant protection through the APRMs and a certain number of the LPRMs must be functioning. In 1100 MWE reactor there are six APRM channels, each of which can average the output from 24 different LPRMs. A minimum of 14 LPRM inputs from at least two axial levels are required or an automatic APRM inoperative trip is generated. The normal number of LPRMs that provide APRM inputs are 22 to 23.

Individual failed LPRM detectors can be bypassed, and neutron flux information for a failed chamber can be interpolated from nearby detectors. A substitute reading for a failed chamber can be derived from an octant-symmetric detector, or an actual flux indication can be obtained by inserting a TIP to the failed detector position.

Inoperative or grossly inaccurate detectors can create operating problems in the short, near and long term situations. As more LPRMs become inoperative; the inferre power distribution becomes more approximate. The uncertainty cascades through all the items derived from the power distribution; e.g., exposure distribution, void history, etc., and eventually can affect the off-line analyses being performed in strategy evaluations. During reactor operation, LPRMs can drift or otherwise give inaccurate signals especially if operating conditions have changed significantly since the last TIP calculation. A start-up or re-start is a good example of changing reactor conditions and hence changing axial flux shapes. During situations such as this, the LPRMs can be indicating local flux levels higher than actually present and consequently prevent further control rod withdrawal through the RBM system (see Section 2.1.1).

3.1.3 Detector Accuracy

In a BWR, the fixed in-core detectors are periodically intercalibrated and at the time of calibration these LPRM signals, relative to each other, are as good as the TIP signals used for the calibration. In some instances, it has been observed that symmetrical TIP locations do exhibit differences even where all reactor conditions are symmetrical. The reasons have been attributed to a variety of causes including: instrument tube bowing, inlet enthalpy non-uniformity, exposure gradients across the fuel assembly, etc. Whatever the cause, the result is that the power distribution inferred using symmetry considerations, is less accurate.

The instrument output is affected by flux gradients. As described earlier, the detector is located in a water gap at the corner of four assemblies. There is a flux gradient along this water channel and the displacement of the instrument tube from its design locations can impact the detector output. The flux gradient in the gap and hence, the flux at the instrument tube can also be affected by bowing of the zirconium cans that house the fuel assemblies. If these cans bow inward, the water gap gets larger and the flux in the gap increases, if can bows outward the gap gets smaller and the flux at the detector decreases.

As stated in 2.1.3, G.E. claims a 2.3% and 5% uncertainty for initial and reload cores respectively due to geometrical effects. In addition, an uncertainty of 3.5% and $\sim$7% for initial and reload cores has been associated with process computer modeling. Again, it has been the authors experience that in some cores the uncertainty is much larger in these values.

Depletion of U-235 in the detector can effect the signals. However, since the detectors are re-calibrated at minimum of a full power month, it is only a consideration in the interim. Factors and algorithms are programmed into the process computer to correct the readings between calibrations.

3.1.4 Time Response

The response of the fission detectors in a BWR is instantaneous. Hence, local flux changes due to rod movement or gross changes due to Xenon redistribution are detected as they occur. The inaccuracies or uncertainties in time are in the overall power distribution inferred from the LPRMs due to the obsolescence of the TIP shape that is the basis of the extrapolation and interpolation procedure.

3.2 Westinghouse

3.2.1 Detectors

The Westinghouse in-core monitoring system consists of movable in-core detectors. There are no fixed neutron detectors in the core. There are thermocouples at the top of selected fuel assemblies. In 15x15 pin and 17x17 pin fuel assemblies the instrument tube is in the geometric center of the assembly. In earlier 14x14 fuel assemblies, the instrument tube was off the geometric center. A typical X-Y positioning of monitored fuel assemblies in the core was described in Section 2.2.1.1 and shown in Figures 2.2.3 and 2.2.4. This selection of fuel assemblies with instrument tubes that can be accessed by a movable detector is such that when rotated into one octant all but the center assemblies are "monitored" through symmetry considerations.

The power distribution in a Westinghouse PWR is determined from the movable in-core detectors. The detectors provide a current which is proportional to the U-235 fission rate at the detector location as it traverses the fuel assembly. There are typically 4 of these movable detectors and each one can access some common instrument tubes. The movable detectors are inter-calibrated by sequential insertion into common fuel assemblies. In this manner, the traverses for all the fuel assemblies capable of accepting a detector are normalized relative to each other.

The data obtained from this in-core map is sent off-site to be converted to a power distribution. A calculation procedure is used to infer the relative power in the monitored assemblies from the fission rate at the detector location. This procedure is given in some detail in Section 4.0, and summarized briefly here. The cross calibrated values from the movable in-core detectors are input to a computer code. The code has pre-calculated factors which relate the instrument fission rate to the average power in the assembly where that particular traverse was taken. This inferred power is compared with predicted and through an algorithm the predicted power in unmonitored assemblies are raised or lowered accordingly. This combination of predicted and measured data thus results in a "measured" power distribution throughout the core.

The accuracy of this power distribution is dependent on a number of factors. For example:

- If the core is not in equilibrium because of changing condition, flux shapes can change during the time it takes to map the whole core, and hence the traverse data is not cross-consistent from assembly to assembly;
- There can be drift in the movable detectors which may go undetected if the normalization traverses are only done once and the map takes a long time;
- The conversion of the detector fission rate to assembly power is an analytic procedure which is dependent upon the isotopic content of the assembly, the presence or absence of control rods, the core location history of the assembly, etc;
- The adjustment of the predicted power in the unmonitored assemblies is based on apriori assumption as to the behavior of these assemblies relative to the predicted/inferred power of the monitored assembly.

These later two effects are discussed in more detail in Section 4.0.

3.2.2 Reliability and Impact of Failure

As stated at the beginning of Sec. 2.2.1.1, the in-core monitoring system provides no operational plant control. The only requirement is that a minimum of two thimbles (instrument tubes) be available in each quadrant for mapping purposes. However, implicit in the requirements of a monthly core map and calibration of ex-core instruments is the requirement that if tilt is indicated by the ex-cores a movable detector be available to perform a map.

If all the movable detectors were inoperative for any length of time, sooner or later the need for an in-core map would affect operations. Similarily, the fewer the number of thimbles available for traversing, the more uncertain are the algorithms for converting the fission rate to assembly power and the less accurate is the overall power distribution. This can reflect back on operations through axial offsets and quadrant tilts that are not representative of the actual core conditions.

3.2.3 Detector Accuracy

In a Westinghouse PWR, the movable in-core detectors are inter-calibrated at the time a core map is performed. It has been observed in some instances that these in-core detectors tend to drift during the time it takes to perform a core map. This has been alleviated by intercalibrating these detectors at the start of the map and at the end of the map (some utilities have performed the inter-calibration in the middle of the map as well). The observed asymmetries in fuel assemblies that are geometrically similar has been on the order of plus or minus a few percent. Since the monitoring is performed on a quadrant basis, these individual asymmetries (or inaccuracies) tend to disappear in the average as far as tilt is concerned. However, they do show up in the power distribution and the dependent parameters.

Flux gradients can have a significant impact on the inference of a power distribution in a PWR. These gradients occur from a number of causes. The assemblies on the periphery of the core experience a sharp flux gradient by a neutron leakage from the assembly. This results in two effects insofar as the in-core monitoring is concerned.

- One, care must be taken in obtaining the factors for converting detector indication to power in these peripheral assemblies.

- Two, the flux gradient results in an exposure gradient that is built in during the residence of the assembly on the periphery. When the assembly is relocated toward the interior of the core, this exposure gradient still exists and can cause a flux gradient in the assembly.

- An additional gradient effect is the presence of control rods. These perturb the flux in space and energy such that if they are present for any significant time in one cycle, it can affect the detector/fuel assembly relation in the next cycle.

- In 14x14 pin assemblies where the instrument tube is located off-center, the orientation of the assembly, especially for peripheral assemblies, must be considered in interpreting the instrument signal.

In the W reactors, as in any PWR, the effect of flux gradients must be addressed in the analyses done to generate the factors that are used to infer relative power from detector signals. (Section 4.0.)

3.2.4 Time Response

The response of the fission detector in a Westinghouse PWR is instantaneous, providing the detector is in the core. The ability to track local flux changes due to rod movement or xenon redistribution is affected more by the capability to perform continuous maps than it is by detector response.

3.3 Babcock and Wilcox

3.3.1 Detectors

The Babcock and Wilcox basic in-core monitoring system consists of fixed in-core detectors, as described in Section 2.3. There can also be thermocouples at the top of selected fuel assemblies. The instrument tube is located at the center of an assembly. As described in Section 2.0, there are typically 52 assemblies (the actual number varies from reactor to reactor, depending upon core size) with the in-core detectors at 7 axial locations; there are 16 symmetrical detectors at each of the 7 axial levels, four in each quadrant. These are illustrated in Fig. 2.3.2.

The power distribution in a Babcock and Wilcox core is determined from the fixed in-core detectors. The detectors provide a current which is proportional to the rhodium absorption rate at the detector location. The detectors are designed with the objective that the proportionality constant is the same for all detectors and hence no intercalibration is provided for; the background is accounted for separately for each detector.

The instrument signals are converted to the relative power in the fuel assembly surrounding the detector through a calculational procedure. This procedure is given in some detail in Section 4.0 and is briefly summarized here. The symmetry monitors are used to determine a quadrant "tilt" factor. The detectors, when rotated into a quadrant, represent every type of fuel assembly in the core, and assuming 1/8th core symmetry, all assemblies are monitored. As the detectors are rotated into each quadrant they are corrected for the tilt in that quadrant by the tilt factor. These corrected values are converted to relative power in the assembly segment or node surrounding the detector by a series of factors that account for moderator temperature, fuel exposure, rodded or unrodded, soluble boron concentration, detector exposure, etc. This results in the relative power in each assembly at 7 different axial locations.

The accuracy of this power distribution is dependent on a number of factors.

- Implicit in the processing of the signals is the assumption that the detector readings are absolute; i.e., that variations in symmetrical detectors are due to the variations in the neutron flux and not the detector itself. The evidence to date places some doubt on the validity of that assumption.
- The conversion of the rhodium reaction rate to surrounding assembly power is an analytic procedure and the results may be subject to the model used to generate the factors. Not only is it dependent upon the environment around the detector (i.e., assembly exposure, PPM, rods, etc.) and the detector history, but there is some evidence that the neutron group structure used has an effect due to the large rhodium resource at ∿lev. Further, comparing the rhodium cross sections quoted by two vendors using rhodium detectors show differences which could affect the interpretation of the signals.

3.3.2 Reliability and Impact of Failure

As stated in Section 2.3.1.1, the in-core monitoring system provides no direct operational plant control. However, the in-core system does provide the function of assisting in the periodic calibration of the out-of-core detectors in regard to core imbalance trip limits.

In a typical large B&W reactor there are 52 in-core flux detector assemblies with 7 axially spaced detectors per assembly for a total of 364 individual detectors. Above 80% power at least 23 individual in-core detectors must be operable to assist in the periodic calibration of ex-cores.

The minimum requirement for 23 individual in-core detectors is based on the following:

a. An adequate axial imbalance indication can be obtained with 9 individual detectors. Figure 3.3.1 shows a typical set of three detector strings with 3 detectors per string that will indicate an axial imbalance that is within 8 percent (calculated) of the real core imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

b. Figure 3.3.1 also shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

c. Figure 3.3.1 shows a combinations of a and b to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

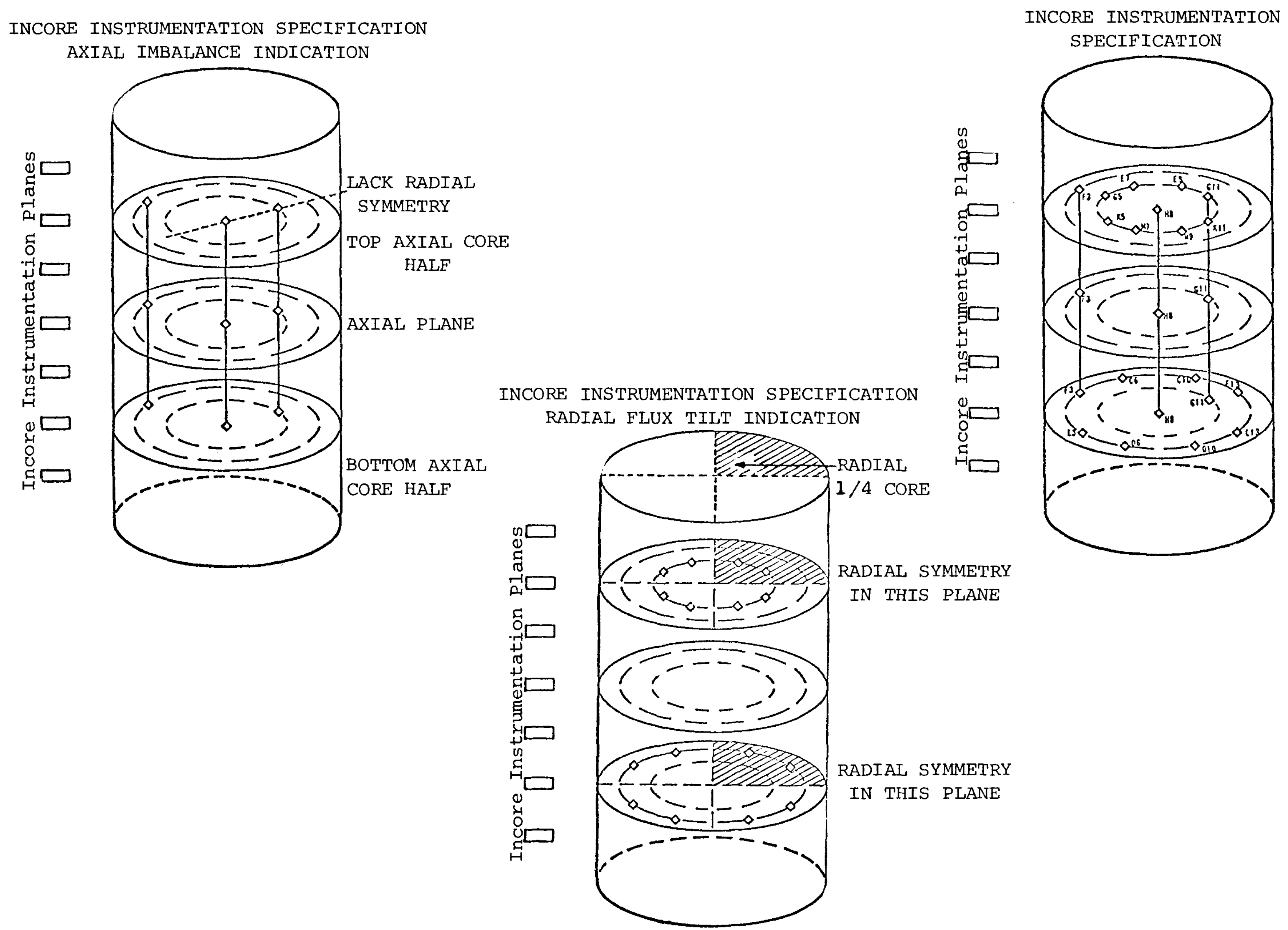

INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION
Incore Instrumentation Planes
LACK RADIAL SYMMETRY
TOP AXIAL CORE HALF
AXIAL PLANE
BOTTOM AXIAL CORE HALF
INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX TILT INDICATION
Incore Instrumentation Planes
RADIAL 1/4 CORE
RADIAL SYMMETRY IN THIS PLANE
RADIAL SYMMETRY IN THIS PLANE
INCORE INSTRUMENTATION SPECIFICATION
Incore Instrumentation Planes

FIGURE 3.3.1

As stated, at least 23 in-core detectors must be operable to check power distribution above 80 percent power. These in-core detectors may be read out either on the computer or on a recorder. If a set of 23 detectors in specified locations is not operable, power will be decreased to or below 80 percent.

To date, the fraction of inoperative detectors appears to be small so that meeting the above requirements has not been a problem.

3.3.3 Detector Accuracy

The fixed in-core detectors in a B&W core are self-powered neutron detectors with rhodium being the emitter and, as stated earlier, there is no cross calibration. According to B&W, manufacturing tolerances associated with the detector mass of the self-powered neutron detector contribute an error of not more than 1% in the detector's output signal. Following installation, the in-core detector system is capable of measuring the relative flux to within 5% of the true flux when used in conjunction with an adjacent background detector. B&W believes that use of the computer-calculated correction factors will limit additional inaccuracies to no more than 1% per year for average flux conditions. These tolerances are necessary to ensure meaningful data on reactor power distribution. Experience with these detectors to date does not indicate this degree of accuracy. The evidence is that symmetrically located detectors can give significantly different outputs, not attributable to any real asymmetry. This, of course, can get reflected into the inferred power distribution and all the related quantities such that the validity of the results are questionable.

Flux gradients can have a significant impact on the inference of a power distribution in a PWR. These gradients occur from a number of causes. The assemblies on the periphery of the core experience a sharp flux gradient caused by a neutron leakage from the assembly. This results in two effects insofar as the in-core monitoring is concerned. One, care must be taken in obtaining the factors for converting detector indication to power in these peripheral assemblies. Two, the flux gradient results in an exposure gradient that is built in during the residence of the assembly on the periphery. When the assembly is relocated toward the interior of the core, this exposure gradient still exists and can cause a flux gradient in the assembly. Another cause of gradient is the presence of control rods which cause a serious perturbation in the spatial flux in the assembly and effect the short-term and long-term considerations for instrument interpretation. In particular, there is some evidence that the interpretation of rhodium detector signals is sensitive to the neutron group structure and fuel assembly spatial configuration used in generating the conversion factors when control rods are present. This is discussed further in Section 4.0.

The detector signals must be compensated for the depletion of rhodium. At the beginning of detector life, a 100,000-ohm resistor, in series with the local flux detector signal, produces a 50-millivolt input signal from the average core neutron flux at full power. The detector signals must be compensated for the burnup of the neutron-sensitive material. The computer system integrates each detector output current and generates a burnup correction factor to be applied to each detector signal before it prints out the corrected signal in terms of percentage of full power.

In series with the plant computer input signal-conditioning resistor is a variable in-core monitoring recorder input signal-conditioning resistor. The variable resistor can be adjusted periodically to compensate for the burnup of rhodium. Since rhodium burnup in the average core flux at full power is less than 1.5% per month, monthly adjustment of the variable resistor will limit readout errors to less than 1.5%.

3.3.4 Time Response

The nature of the activation-type neutron detector involves an inherent time constant in the detector response which is related to the half-lives of the decay. In the case of rhodium, two half-lives are involved: 4.4 minutes for the isomeric state and 42 seconds for the ground state of rhodium-104. The primary source of the detector signal is from the 2.4-MeV beta decay from the 138-barn ground state rhodium-104, which has a characteristic half-life of 42 seconds. The lower-energy beta decays represent less than 2% of the total signal. There is some production of an excited state of rhodium-103, but because it represents less than 1% of the total decay, this fraction can be neglected. The 4.4-minute half-life of the 12-barn isomeric state rhodium-104 results in a residual current that remains after a rapid shutdown.

Figure 3.3.2 shows the response of the rhodium to a step change in reactor neutron flux. It shows graphically why these detectors are not satisfactory for control or protection system inputs. However, the response is sufficient for monitoring slow transients such as a Xenon oscillation.

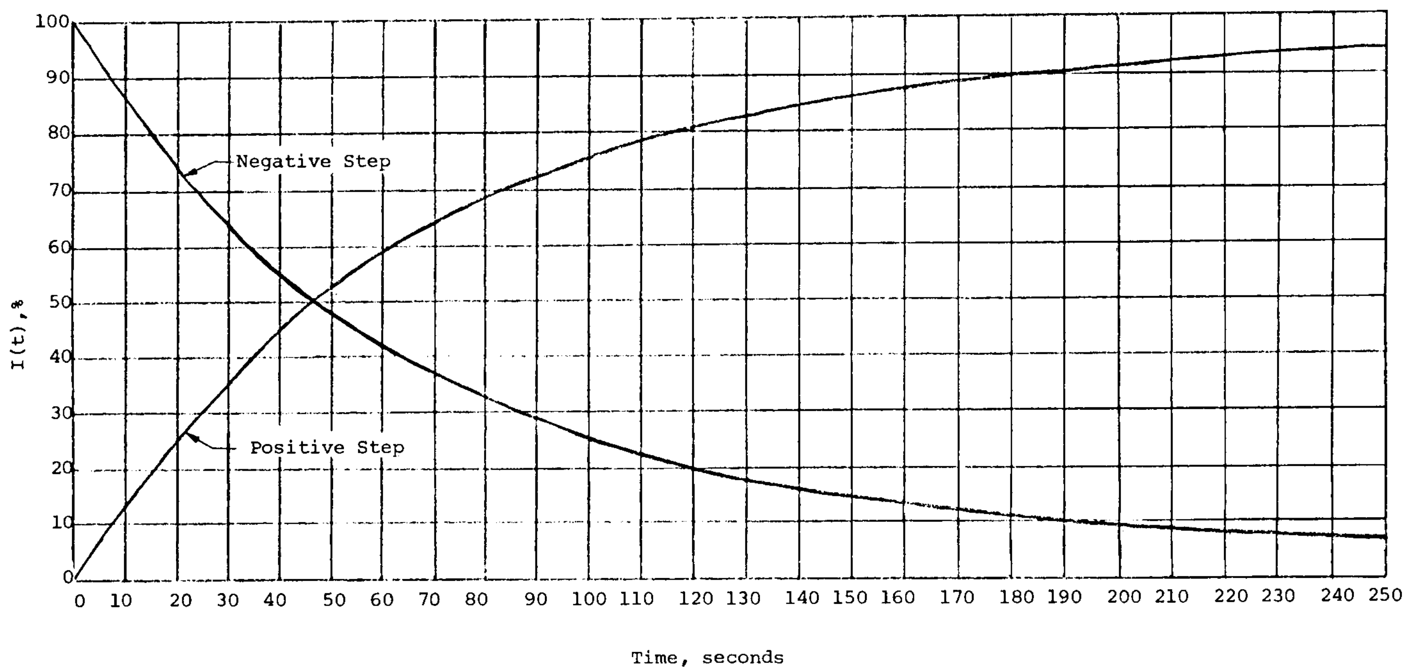


FIGURE 3.3.2

RESPONSE OF RHODIUM DETECTOR TO STEP FLUX CHANGE

3.4 Combustion Engineering

3.4.1 Detectors

The Combustion Engineering in-core monitoring system consists basically of fixed in-core detectors, although (Section 2.0) there is some limited use of movable detectors. There are also thermocouples at the top of selected fuel assemblies. The instrument tube is located at the center of the assembly. There are four detectors in each instrumented assembly with their centers located at 20%, 40%, 60% and 80% of the core height. In a typical C.E. reactor which started producing power in early 1976, there are 45 instrumented assemblies out of a total of 215 assemblies in the core. The positioning of the detectors in the core, both axially and in the X-Y plane, was described in Section 2 and illustrated in Figure 2.4.3 . This arrangement, assuming octant symmetry, covers about 50% of the assemblies in an octant.

The power distribution in a C.E. core is determined from the fixed in-core detectors. The detectors provide a current which is proportional to the rhodium absorption rate in the detector. The design objective of the detectors is that with appropriate corrections, such as for background and depletion, the proportionality constant is the same for all detectors. The data from these detectors is periodically sent off-site to be converted to a power distribution. A calculational procedure is used to infer the relative power in all assemblies from the rhodium reaction rate at the selected locations.

In essence, the method of performing the instrument signal analysis is similar to the synthesis techniques frequently used to approximate solutions to three-dimensional problems. The signals from instruments along one channel are used to infer an average axial power shape in the surrounding fuel assembly. These shapes are then used in conjunction with a library of coupling coefficients to obtain the corresponding power profiles in fuel assemblies which are uninstrumented or contain inoperable detectors. These coupling coefficients are allowed to vary axially over the various vertical regions defined by differing control rod patterns. Additional libraries, derived from standard reactor analysis calculations, are then employed to obtain the peak power and hot channel heat flux in each assembly.

The accuracy of this power distribution is dependent upon a number of factors. For example, quadrant symmetry is a basic premise in generating a power distribution (the amplitude and orientation of a tilt is obtained by comparing symmetrically located detectors but does not appear to be factored in the power distribution). Inferring a power shape in an assembly from 4, 40 cm long detectors and extrapolating this shape to unmonitored assemblies is a complicated analytical procedure which depends upon the environment around the detector (rodded, exposure, etc.), the detector history, and the applicability and accuracy of the detailed apriori calculations done to generate the data base used in the procedure. This is discussed further in Section 4.0.

3.4.2 Reliability and Impact of Failure

The in-core monitoring system provides an auxiliary source of information concerning the gross power distribution in the core; however, it assumes no role in the protection of the plant and is not necessary for economic operation of the reactor, per C.E. position. However, according to Tech Specs, the in-core monitoring system may be used to monitor the core power distribution, i.e., linear heat rate.

There are restrictions on operations that are tied to the in-core monitoring system. For example, in one typical plant, it is required that 75% of the in-core strings be operable. An operable string is defined as having at least 3 detectors operable. Also, of this 75%, there must be a minimum of 2 quadrant symmetrical strings operable in each quadrant. The requirement is that each operable detector in the operable symmetrical strings (on the same level) must be within 5% of the average.

3.4.3 Detector Accuracy

The fixed in-core detectors of C.E. have little or no cross calibration capability. As in the case of all rhodium detectors, a number of corrections must be applied to obtain a "true" reading. These include corrections for: detector length; spectral conversion factor; termination resistance; background signal correction; and detector burnup correction. The design objective is that after these corrections have been applied, the signal is representative of the actual flux to 5%.

Flux gradients can have a significant impact on the inference of a power distribution in a PWR. These gradients occur from a number of causes. The assemblies on the periphery of the core experience a sharp flux gradient caused by a neutron leakage from the assembly. This results in two effects insofar as the in-core monitoring is concerned.

- One, care must be taken in obtaining the factors for converting detector indication to power in these peripheral assemblies.

- Two, the flux gradient results in an exposure gradient that is built in during the residence of the assembly on the periphery. When the assembly is relocated toward the interior of the core, this exposure gradient still exists and can cause a flux gradient in the assembly.

- An additional gradient effect is the presence of control rods. These perturb the flux in space and energy such that if they are present for any significant time in one cycle, it can affect the detector/fuel assembly relation in the next cycle.

- With detectors 40 cm long, control rods in adjacent assemblies can cause significant axial gradients such that care needs to be exercised in interpretating detector output and converting 4 readings to an axial shape.

In the C.E. reactors, as in any PWR, the effect of flux gradients must be addressed in the analyses done to generate the factors that are used to infer relative power from detector signals. (Section 4.0)

3.4.4 Time Response

As discussed earlier, there is an inherent time constant involved in the detector response which is related to the half-lives of the decay. Section 2.4.1 gives the details of this decay. The time response of the rhodium to a step change in neutron flux was shown in Figure 3.3.2. As can be seen, this type response prohibits use of these detectors for protection systems or fast control. However, the response is adequate for monitoring longer time dependent phenomena such as xenon transients.

4.0 ANALYSES REQUIREMENTS

The previous sections dealt with the need for measurements (Section 1.0), the instrument description (Section 2.0), and the acquisition of data (Section 3.0). This section is directed towards the analyses necessary to interpret these measurements. The discussion is qualitative with the objective of pointing out the difficulties and uncertainties involved. Since the considerations in interpreting the detector outputs, i.e., the types of on-line and off-line analyses needed, assumptions made, and techniques for validating results, are not significantly different among the four vendors, this section treats the four more intercomparatively than Sections 2.0 and 3.0.

As described in Section 1.0, there is a continuous flow of data from the in-core monitoring system which is interpreted, evaluated and the resulting information factored into short, near and long term considerations. The type of information required and why were defined for each of these time categories in Tables 1.1 through 1.6. In addition, this information flow is shown schematically in Fig. 1.1 and Fig. 1.2 for BWR and PWR reactors respectively and its general use and applications is shown in Fig. 1.3.

The core parameters that are of interest to the utility engineers, either operations or design, (and to the vendor designers as well) are not directly measurable. These include such items as: total core reactivity, reactivity worth of control rods, nodal power distribution, peak pin powers, and all the history items from the power distribution; e.g., exposure, rod history, etc. What can be measured directly is a current which is proportional to a reaction rate at a very local position in the core. In a BWR, the current is produced by a miniature fission chamber located at the corner of four fuel assemblies, as shown in Figure 2.1.4. In a PWR, the current is produced by a detector located in the center of the fuel assembly (for 15x15 and 17x17 assemblies) as shown in Figure 2.2.2. As described earlier, Westinghouse uses a small, movable fission chamber, Babcock and Wilcox uses a rhodium detector $\sim$13 cm long and Combustion Engineering uses a rhodium detector $\sim$40 cm long. In all cases the current is generated by a reaction rate at the detector location; i.e.,

$$\text{Current } \alpha \int_E \int_V N\, \sigma(E)\, \phi\, (E,V)\, dEdV$$

N	=	number density of U-235 or rhodium
V	=	detector volume
$\sigma_i(E)$	=	cross section of U-235 or rhodium
$\phi(E,V)$	=	neutron flux at detector

If these local measurements can be converted into a power distribution, all other items of interest will follow. One problem then is to infer the fission rate (or power) in segments of the fuel assembly(s) adjacent to the detector from this local measurement.

The points or locations throughout the core at which these local reaction rates are measured are limited. Section 2.0 showed that approximately 25% of potential instrument tube locations are used, i.e., that instrument tubes are adjacent to or in only 25% of assemblies. In addition, when axial fixed in-core monitors are used; e.g., by B&W, C.E., G.E., there are only 7,4,4 axial locations respectively where these reaction rates are measured. It shows only a small number of locations in the core with detectors. If traversing monitors are available; e.g., G.E., W, and to a limited degree C.E., the reaction rate can be obtained at many axial locations, but again, only in a few radial positions. However, the frequency that the traversing monitors are used, approximately one a month, tends to offset the advantage over the fixed detectors of having multiple axial points. A second common problem then is to infer the fission rate (or power) in segments of fuel assemblies that are not adjacent to any detectors.

4.1 On-Line Calculational Procedure

The starting point for obtaining a "measured in-core" power distribution is the reaction rate at a few local points in the core.* These reaction rates are converted to the relative power in the axial segment of the assemblies near the detector by off-line precalculated factors. These factors are a function of a number of conditions or rather must account for a number of effects; e.g., detector, type of detector, fuel assembly exposure, rodded history, current presence of rods, presence of burnable poisons, moderator density or temperature, fuel assembly location history (gradients), etc. The unmonitored locations in the core are obtained by algorithms which interpolate and extrapolate the monitored values both axially and in the X-Y plane, based on off-line analyses. The final "measured" power distribution which results is an inferred result which was obtained by combining local measurements with pre-determined factors and algorithms.

Let's examine this general procedure in more detail and to facilitate the discussion, separate the analyses into the two categories.

1. Conversion of detector output to power in adjacent fuel segments,
2. Extrapolation to fuel segments without detectors.

Specifics of the calculational procedure are proprietary to some vendors. However, sufficient information is available or can be logically deduced to provide an understanding of what is done even though some of the details may not be exact.

Starting with the PWR's, the basic relationship in the first category is the ratio of detector current (i.e., reaction rate) to the power produced in the fuel segment surrounding the detector. For example,

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*Many of the PWR's have thermocouples at the top of selected fuel assemblies, however, these appear to be used very little in the process of inferring a power distribution and hence are not included in the following discussion.

$$\text{Segment Power } \alpha \text{ f (E,T,PPM,---)*Detector current}$$

where f is a function of as many as nine variables. In most cases these variables can be treated separately, and

$$f(E,T,PPM,---) = f(E)*f(T)*f(PPM)*f(ROD)*f(POWER)*f(LOCATION)*---$$

where f(E) = assembly segment exposure correction factor

f(T) = segment moderator correction factor

f(PPM) = soluable boron correction factor

f(ROD) = correction factor for presence of control rod in segment

f(POWER) = correction factor for different power levels of segment

f(LOCATION) = correction factor for location (past and present) in large gradient regions

Each of these factors is usually represented by quadratic and cubic curve fits (or table sets) which span the limits of the particular parameter variation. The factors are generated off-line prior to when they are needed, by reactor design methods and codes. The factors must be generated for each fuel type in the core; a fuel type being characterized by any parameter which makes it unique, such as enrichment, presence of burnable poison rods, geometric configuration and for some of the X-Y locations in the core.

The second category, extrapolating to fuel segments without detectors, is approached somewhat differently by the three PWR vendors. These are summarized below.

C.E. constructs pseudo detector outputs, on-line, in the monitored segments by assuming these to be proportional to the detector outputs in symmetrical locations. The axial power distribution in these pseudo-monitored and monitored assemblies is obtained by a fitting procedure. This procedure employs a series formed from the first three Fourier mode shapes, cos θ, sin 2θ, cos 3θ. The power shape in the top half of the core is obtained by fitting the upper three detectors and the shape in the bottom half by fitting the lower three instruments. C.E. states that this causes a small discontinuity at the core mid-plane, but this is small and does not influence the accuracy of the generation of the axial peak. In the assemblies that are not instrumented and not symmetrical to any monitored assembly (25% of all assemblies) coupling coefficients which relate the power in the monitored assemblies to the power in adjacent unmonitored are used to obtain their axial power profile. These coupling coefficients are spatially dependent and vary with fuel burnup and control rod configuration.

$$P_{\text{node}}^{\text{adjacent unmonitored}} = P_{\text{node}}^{\text{monitored}} * C(E,L,R)$$

where E = exposure

L = location in the core

R = rodded or unrodded condition

It is pointed out that no assembly can accept a control rod and the in-core detectors simultaneously. Hence, the power shape and level in all rodded assemblies must be obtained from coupling coefficients. After the power distribution has been inferred, another series of factors are used to determine peak pin power or maximum KW/ft. These also depend upon assembly type and location, exposure, rods, etc. Finally, adjustment for quadrant tilt is made based upon the comparison of symmetrically located detectors. These calculations are done off line.

B&W also used the pseudo detector approach, i.e., uses the output of monitored assemblies in symmetrically located unmonitored assemblies. In this case, however, for every unmonitored assembly, there is a corresponding monitored assembly (using octant symmetry). B&W using 7 axial nodes corresponding to the 7 axially located detectors and hence, there is no axial extrapolation performed. When the monitored assemblies are rotated to a particular quadrant, tilt factors which are obtained from comparisons of symmetrically located detectors are applied to the quadrant values of fuel segment or nodal powers. The total peak, pin powers, KW/ ft., etc. are obtained in the same manner as C.E., i.e., applying a set of pre-calculated factors which are dependent upon fuel type, position exposure, rod presence, etc.

Westinghouse's approach is somewhat different. Since the detectors are movable and can provide as many axial values as desired, no axial interpolation or extrapolation is necessary in the monitored assemblies.

The unmonitored assemblies values are obtained in some instances using a symmetry assumption and using monitored values in unmonitored quadrant symmetrical assemblies. Where symmetry is not employed, the unmonitored assemblies values are obtained from off-line 1/4 core predictive analyses that were done for conditions similar to the actual. These calculated values are adjusted up or down according to algorithms which utilize the comparison of calculated and measured reaction rates in the monitored assemblies. As in the other PWRs, the peak power, pin power and KW/ft, etc., are obtained by applying additional pre-calculated factors that are a function of location, exposure, and rod presence.

The BWR approach is similar to those just described. In the case of the first category, conversion of detector output, the basic difference is that the output is related or proportional to the four fuel assembly segments or nodes adjacent to the detector location. The relation is then

$$(\text{Segment Power})_i \; \alpha \; f_i(\text{E,V,ROD---})*\text{Detector Current}$$

where

$i = 1$ to 4, representing one of the segments

and

$$f_i(\text{E,V,ROD---}) = f_i(\text{E})*f_i(\text{V})*f_i(\text{ROD})*\text{---}$$

where

$f_i(\text{E})$	=	exposure correction factor for each of the four segments
$f_i(\text{V})$	=	void correction
$f_i(\text{ROD})$	=	correction factor for any combination of four rods present
$f_i(\text{VH})$	=	void history correction factor
$f_i(\text{RH})$	=	rod history correction factor

(There are also adjustments made to the assembly of interest which are dependent upon the void and exposure of the other three assemblies around the instrument tube, but does not consider directly any other adjacent assemblies.) These factors can be represented by cubic or quadratic curve fits (or table sets) and can sometimes be combined. They are generated off-line by reactor design methods and codes (Section 4.2). They must be generated for each fuel type in the core; a fuel type being uniquely characterized by enrichment, burnable poison placement, burnable poison loading, assembly geometry, etc. These off-line calculations must be performed over a sufficient range of voids, exposure, roddedness, etc., to encompass the conditions the segment will encounter during its life in the core.

In the second category, extrapolation to unmonitored segments, the G.E. approach is to construct pseudo detector outputs in the unmonitored locations that are geometrically symmetrical to monitored locations. Assuming quadrant symmetry (or correcting the detector output for non-symmetrical configurations) approximately 90% of the asemblies are adjacent to LPRM strings, either real or pseudo. The four LPRM outputs in each string are used to construct 24 axial points. This is done by utilizing the TIP shape for that string location that was stored the last time a TIP map was made. (If conditions have changed, an individual TIP can be run to improve the shape of a particular LPRM string, or a whole core TIP map may be necessary if the change was significant and was reflected core wide). With the LPRMs converted into 24 values, the power in the adjacent assembly segments is obtained using the above described factors. A fraction of the total power is assigned to the 10% unmonitored assemblies (based on off-line calculations) and a core power distribution obtained. This power distribution is then used to calculate a void distribution for each fuel assembly. The new void distribution is used to obtain new detector conversion factors (which are void dependent) and the power distribution is regenerated. This continues until convergence. As in the case of the PWRs, pin powers, peak powers, KW/ft, etc., are obtained by applying another factor or factors which relate nodal or segment power to peak pin power in the segment. These factors are generated off-line and also are functions of fuel type, exposure, void, rod presence, etc.

In each of the discussions above, the starting point was a validated detector signal. In actual practice the signal must first be cross-calibrated (G.E., W) and/or corrected for depletion of detector material (B&W, C.E. and G.E.), background (B&W, C.E.) and spectrum (C.E.). These corrections also utilize parameters generated off line.

In summary, each of the vendors use the same basic approach to convert signals to assembly power; i.e., the corrected or cross-calibrated detector reading is converted to assembly power by pre-calculated factors and then these results are extrapolated and interpolated to the nonmonitored segments through pre-calculated factors and/or algorithms.

Once the power distribution has been generated (whether PWR or BWR) all items needed for short, near and long term requirements can be obtained. (These are defined and described in Tables 1.1 through 1.6). Included are the items necessary to assure compliance with limits imposed from safety considerations (Tech Specs) and operating requirements. It is apparent that the procedures for inferring the power distribution are by necessity simplified and unless the factors utilized provide a power distribution that is representative of the actual, a decreased plant capacity factor could be experienced.

4.2 Off-Line Generated Parameters

As has been mentioned numerous times, off-line (meaning independent of measurement) predictions are necessary to determine the constants and factors for converting the detector reading to assembly power. The calculations of these factors are not necessarily straightforward and the assumptions in the various procedures or approaches can lead to variances in the inferred power distribution.

The detailed description of the procedures, assumptions and models used in the off-line calculations warrants a separate report and is not in the present scope. However, a summary description will help to give an appreciation of this phase of the in-core power distribution measurements.

The determination of these factors just described requires a detailed two-dimensional, multi-neutron energy group calculation. In the case of the PWRs this is normally a 1/4-core two-dimensional diffusion theory calculation (PDQ) where the ratio of the activation rate at the detector to the relative power in the surrounding assembly can be edited. Multiple calculations must be done in order to obtain the effects of assembly exposure, control rod presence, burnable poison presence, moderator temperature, soluble boron concentration, etc. The latter two are second order effects and can be treated often by extrapolation or branch calculations. Sufficient spatial detail is necessary to account for off-center instrument tubes (14x14 pin fuel), the sharp flux gradients that occur in assemblies on the periphery of the core and the presence of control rods (or burnable poisons) in the instrumented and/or neighboring assemblies. This detail must be able to preserve the isotopic composition across an assembly such that when it is moved from the periphery to the interior of the core, the flux at the detector and the flux in the assembly represent actual conditions experienced in the reactor. In addition, consideration must be given to the mode of operation; e.g., if control rods are in the core a portion of the time, then sufficient calculations must be done not only to represent the rod presence but to represent the residual effect of the rod after it has been withdrawn. (See Section 4.3.)

In the case of the BWR, the calculations are normally done for single assembly or for a cluster of assemblies.
Inherent in this approach is the assumption of an infinite array of such assemblies, which can lead to factors that may not be representative, especially if the single assembly or cluster is bounded in the actual core by assembly significantly different in design or history. The analyses must consider the presence or absence of control rods, and due to the instrument location (Figure 2.1.4). the presence of any one of four control rods can effect the detector flux to assembly power relation in each of four assemblies. Consideration must be given to the operating mode and the effects of a control rod having been present (and causing a severe flux and hence exposure gradient) for part of the time. The effect of voids must be considered, these cause a change in the spatial-spectral relationship between the assembly flux and detector flux. The U-238, U-235, Pu contributions to power can vary due to changes in the neutron spectrum even though the detector reading remains constant; e.g., the power produced in a node at 60% void and a node at 0 void is not the same even if the detectors give the same output. Also, void history affect the isotopic concentration and hence the factor relating detector reading to power. The presence or absence of burnable poisons and their affect on the relationship of assembly power to flux at the detector must be accounted for, and this relationship is also a function of voids and exposure.

Summarily, in both the PWR and BWR cases, sufficient calculations must be done to encompass all the different fuel types in the core and all the different conditions these fuel types will experience during the residence in the core.

Some current operating reactors have as many as 12 different fuel types and this coupled with the variable conditions each type experiences, represents a significant off-line effort to support the on-line calculations and assure the factors being used are adequate.

4.3 Effect of Various Operating Conditions

The operating conditions have an effect on the inference of the power distribution insofar as they produce local conditions in the core that have not been addressed in determining the factors that convert detector signals to local power. (Section 4.3 discussed some examples of this.) In a BWR, the local node can and does have a wide range of voids; can be effected by adjacent control rods and even control rods inserted by adjacent assemblies, can be effected by the type of assembly that surrounds it; all which influence the instrument interpretation. For example, current BWR cores can have a variety of fuel designs adjacent to the LPRM and the "state" of each node in the assemblies can represent a variety of conditions, all of which effect the detector interpretation, and the analyses required for the interpretation. All these conditions must be accounted for in the off-line analyses at least to the point of showing them to be second order. In a PWR, operation with control rods present influence the detector interpretation both currently and in the future, as was mentioned earlier. The placement and orientation of the fuel assemblies when they are shuffled in succeeding cycles has an effect on the interpretation of the detector, through the gradients and the assumption in the off-line calculations treating these gradients.

Again, it is a matter of whether or not the analyses that generate the detector conversion factors and the coupling coefficients have encompassed all the operating conditions and resultant local environments that do occur in the core.

4.4 Time Dependency

The various approaches to the measurement, or rather inference, of an in-core power distribution reflects the philosophy of each vendor as to the utilization of the information. This is also present in the time considerations for obtaining an inferred power distribution.

Two vendors, C.E. and W, perform the detailed interpretation of the in-core detectors off-line. There is a time delay involved which could be long (although C.E. is automating the on-site/off-site link). B&W and G.E., on the other hand, do infer a power distribution on-site so that the time involved is the order of minutes. In all reactor types, there are time requirements for determining the power distribution (or some derivative such as linear heat) after an abnormal operating situation, such as a dropped or stuck rod. In the case of the non-automated off-site analyses, it appears that it would be difficult to perform the required analyses in the specified time. It is the author's opinion that the requirements on the frequency of obtaining in-core power distributions in PWR's is going to increase due to elimination of conservatisms associated with ex-core monitoring, due to increased interest in the detailed fuel performance history, and due to the eventual operation of the plants in a load follow mode where the details of the power distribution shifts must be known to assure compliance with operating and licensing constraints for the specific operation conditions.

4.5 Importance of In-Core Power Distribution

At this point it may be worthwhile to review for a moment just what the purpose of this analyses is, that is, why an in-core power distribution is wanted.

Section 1.0 described in some detail the information requirements associated with operating a nuclear power reactor. These requirements were separated into short term, near term and long term categories for definition perspective and discussing purposes. (Tables 1.1 to 1.6) A common bond among these items is that they all are obtained directly or indirectly from the detailed in-core power distribution. This is true whether the requirement is to assure safety (tech spec) or operational limit conformance, or for strategic purposes to plan future operations.

Sections 2.0 and 3.0 described the monitoring systems and addressed some of the implications of failed or erroneous indicating detectors. The implications and consequences of inaccurate interpretation of the detector output, i.e., the analytic conversion of the detector output to a power distribution, are equally severe. For example, if the analytic conversion results in a peak power higher than is actually occuring in the core, an unnecessary constraint on total core power output could be experienced with a resulting loss in capacity. Conversely, if the analytic conversion results in a peak power lower than what is actually occuring, it is possible to have the core operating with peaks that exceed safety or operational limits.

There are longer term consequences than can result from an inaccurately inferred power distribution. Strategies for future operation are based on current power distributions and on off-line predictions of future power distributions.

The latter uses the exposure histories that have been accumulated in the core. If those inferred on line are erroneous, and are used the predictions will be erroneous. If calculated histories are used which are in disagreement with the on-line values, the designer is uncertain as to the validity of his prediction, which can impact his strategy discussions and the degree of conservation that he factors in.

Thus the in-core power distribution is a necessary part of reactor operations and an important and vital part of the "measurement" of the in-core power distribution is the analyses needed to convert the detector signals to power.

Another area that is time affected is the off-line analyses for the conversion factors and constants. As discussed in Section 4.2 and 4.3, these factors are time-dependent and must be periodically updated when core conditions change significantly from those assumed in the calculations generating the constants. Also, each time the core is reloaded, shuffled or otherwise perturbed, a new set of factors can be required.

4.6 Validation of Results

Just as the inference of the in-core power distribution is not a straight forward procedure, the validating of the results are qualitative and not readily amenable to an absolute determination. The problem is that the "measured" power distribution is a function of measured data and analytical interpretation. Both of these are subject to uncertainties which is reflected in the end result, the inferred in-core power distribution.

Examples of approaches that can be used to help validate results are as follows:

a. Comparison of Symmetrical Detectors

All the cores have symmetrically located detectors, (in the case of W, they are movable). (In a BWR, certain normal rod configurations negate this symmetry and hence only detector symmetry exists about half the time.) These symmetrical detector readings can be cross-compared. If different, individual readings can be checked for reasonableness; e.g., do neighboring detectors show the same variation, in this manner the variations can be analyzed and determined if they are core related or detector related.

b. Comparison with Off-Line Predictions

If off-line predictions are being compared to the inferred power distribution (or better, to the actual detector readings), they can be of help in identifying faulty or erroneous detector readings. For example, if the predictions and measurements agree except in isolated cases, the reading may be erroneous. If there is a pattern to the disagreement, then input parameters in the off-line model may be varied in order to attempt to duplicate the observed pattern and to evaluate the magnitude of the variation needed to reproduce the observed phenomena.

c. Comparison with Other Plants

Cross comparing results from other plants can be a valuable means of aiding the verification. It can be determined if the variations in symmetrical detectors are the same from plant to plant; and a statistical assessment can be made on deviations in similar plants. The differences between on-line power inferred distributions and off-line predictions can be examined for trends from plant to plant and assessments of the cause made.

d. Gamma Scans

Longer term measurements on the fuel assemblies after removal from the core can provide data for verification of on-line calculations. γ scans provide indications of the time integrated power distribution and isotopic ratios of can also give some insight on the local exposure distribution.

As can be surmized, the validations of the 1) inferred power distribution and 2) the predicted power distribution are not readily separable and really have to be done in tandem. A detailed knowledge of how the on-line inferred power distribution is obtained and how the off-line predictive calculations are performed is required in order to make assessments as to the validity of one or the other.

4.7 Maintaining Codes for Detector Interpretation

All the vendors employ a complicated procedure for reducing the actual detector output and inferring a complete core power distribution from this output. The procedure includes computer programs (on-line or off-line) to correct the raw detector signal, to convert the corrected signal to nodal power, to extrapolate or interpolate to the nodes and/or assemblies that are not instrumented, as well as the off-line predictive codes that generate the factors and inputs for the interpretative codes.

Among the utilities that the author is familiar with, the involvement in code maintenance, updating and knowledge of the technical content varies widely. Most have the capability for maintaining the software from a programming standpoint and for incorporating new code which are supplied by the vendor. In some particular instances, the utility programmers are more knowledgeable about their own particular on-line computer system than the vendor.

The knowledge of and familiarity with the technical content and bases of the on-line software also varies widely. As a generalization, the depth of understanding by utility engineers decreases from PWR's to BWR's, and within the PWR's from Westinghouse to Babcock and Wilcox and Combustion. This tends to parallel the complexity of the particular monitoring systems. In a few instances (specifically Westinghouse systems), the utilities have provided the factors needed to infer the power distribution from the detector output. This will probably happen more and more. Utilities are acquiring the capability to perform the off-line analyses necessary to generate these factors and are moving into the area of operations support with these off-line models. As they do more in the operating strategy area which includes reload patterns and enrichments, specification, and buy fuel from different vendors, it appears inevitable that they will have to provide the necessary data for on-line monitoring themselves. This will most likely occur in stages, first as an adjunct to the vendor, then in parallel or as a backup to the vendor, and finally as the prime supplier independent of the vendor.

5.0 SUMMARY AND OBSERVATIONS

The "monitoring" of a core is the determination whether or not certain parameters (heat flux, linear kw/ft, etc.) are within the limits set by safety, design, or operational considerations. As described in Section 1.0, all these parameters of interest are derived from the power distribution and hence "monitoring" the core is really assuring that the in-core power distribution at any time for any conditions experienced is within the bounds defined by the safety and operating requirements. The utility operating the plant obtains this assurance from periodic "measurements" of the in-core power distribution. The various systems used for these "measurements" and the means of inferring the in-core power distribution from what is actually measured have been described in Sections 2, 3, and 4, and the flow and utilization of information that is generated from in-core measurements is summarized in Figures 1.1 and 1.2.

The general approach to obtaining a measured power distribution is basically the same for each LWR vendor; i.e., perform a local measurement of a reaction rate using some type of in-core detector, correct the output signal for side effects such as detector depletion, background, etc., convert the detector signal to assembly power by algorithms and/or pre-determined factors which have been previously calculated offline, and extrapolate and interpolate to the locations in the core not neighboring a sensor. The differences from vendor to vendor are in the type of detectors used (Section 2.0), the frequency of measurement (Section 3.0), and the utilization of the information (Section 1.0).

Since the beginning of commercial nuclear power, the trend has been to place more and more emphasis on detailed monitoring of the core, both from a regulatory viewpoint and an operational viewpoint. Now for example, reactors are being operated with administrative constraints on power level and rate of power increase, derived from consideration of the effect on local fuel lifetime. This requires continuous detailed monitoring of the core. It seems predictable that this trend will continue in the future and will continue to impact the operations of the plant. It is expected that these trends will lead to more frequent inference of a detailed power distribution and more direct use of the results. Today the PWR vendors emphasize that the in-core system is for "fuel management" or for "additional" information and is not needed for nor does it impact plant operations, other than an occasional check of ex-core calibration; and the BWR vendor says that the on-line computer is an aid that is not required for operation (and hence the ability to obtain a power distribution rapidly is not required). This was no doubt the original design intent (and may still be) but in actual practice, the utilization of the in-core detector information as a support to reactor operations is becoming commonplace.

The determination or inference of a core power distribution from the in-core detectors is subject to uncertainties that arise from two major areas:

1. The measurement itself, i.e. the detector signal.

2. The analytic conversion of the signals to a meaningful parameter, i.e. power distribution.

Items that effect the measurement are summarized below:

The total exposure the detector has experienced; The current produced is proportional to the number of neutrons absorbed by the detector. As the absorbing material is depleted, the number of absorptions, and hence, the current generated, for a given neutron flux changes.

Manufacturing tolerances; If there are no means of cross-calibrating the in-core detectors (either before or after they are in the core), then each detector must have the same mass of absorbing material in order to produce the same signal in the same neutron flux.

Gamma radiation levels; For some types of detectors, rhodium for example, the gamma radiation field can give rise to a current which is a significant fraction of that produced by neutrons, expecially when the detector has experienced some depletion.

In a BWR the detectors are located outside the assemblies and are sensitive to the characteristics of each of the four assemblies around it. For example, whether or not the assembly(s) are rodded, what the assembly exposures are, what the steam content of each is, what the enrichment distribution in the assembly pins is, all impact the flux at the detector. In addition, there are gaps adjacent to the detector and the signal is sensitive to boiling in this gap, the centering of the detector (or instrument thimble) in the gap, and gap width variations due to fuel assembly can bowing.

In a PWR the detectors are located at or near the center of the assembly. (in 14 x 14 pin fuel the instrument thimble is off center). The detector output can be influenced by the assembly exposure, soluble boron content and moderator temperature, and the presence of burnable poison pins or control rods. Exposure gradients, such as those that occur in assemblies that have resided on the core periphery, also impact the detector flux.

These effects can be accounted for in the analytical conversion of the signal to a nodal or a fuel segment power if one is aware of their presence and magnitude. Unfortunately this is not always known and the consequence can be an inferred power distribution that does not represent the actual core power distribution.

Items that effect the conversion of the signals are summarized briefly below:

The conversion of the in-core detector sigrals to the power in each fuel segment or node in the core is a complicated procedure which involves a number of basic assumptions as well as pre-calculated constants from off-line analyses. The detector strings or tubes are located in or near approximately one quarter of the fuel assemblies; for C.E. and G.E., they are located at four axial locations, and for B&W at 7 axial locations. Hence, the number of points at which a measurement is made are relatively small compared to the number of fuel segments or nodes at which the power inferred. To alleviate this, most approaches assume quadrant symmetry and/or that factors and algorithms can be used to convert the instrument signal in one quadrant to what it would be in a symmetrical location in another quadrant. For the remaining locations, both axially and radially, that are not "directly or indirectly monitored" factors and algorithms are used to extrapolate or interpolate from those locations that are. When quadrant symmetry is not used, then the extrapolation and interpolations scheme must be used for about 75% of the assemblies. Further, if the number of detectors that are inoperative, failed or giving erroneous signals are very large, these extrapolations and interpolations have a large uncertainty and hence so does the inferred power distribution.

The segment or nodal power is obtained by applying a series of factors which convert the current at the detector location to the power in the neighboring node. For example:

$$P_{NODE} = \text{INST. SIGNAL} * f(E) * f(r) * f(PPM) * f(D.E) * f(V) * \ldots$$

where the factors are functions of nodal exposure, moderator temperature, presence of rods, soluble boron, detector exposure, steam void, etc.

It is not possible to perform these off-line calculations for all the combinations of conditions and effect that the various fuel segments or nodes in the core will experience. The usual approach is to perform sufficient analyses to cover the range of conditions expected in the core and during the actual monitoring extrapolate and interpolate to the conditions of interest.

These off-line calculations are also subject to judgments made by the designer in the course of the analyses. For example, the neutron group structure to be used, the number of assemblies to include in each calculation, which effects are second order and can be ignored or approximated, what combinations of exposure, void or ppm to use and so forth.

In summary then, both these off line and on line analyses which are necessary to infer a power distribution from the in-core detectors give rise to uncertainties which are reflected in the final result, "the measured in-core power distribution."

As stated earlier, in a very general sense, the approach to determining the in-core power distribution by each LWR vendor is similar. However, the specifics of the systems used are different. As might be expected each system has advantages and disadvantages and it will be helpful at this point to briefly review some particulars in the various systems before addressing the potential problems that utilities will encounter in the future in the area of "measuring and utilizing" the in-core power distribution.

Westinghouse. The Westinghouse approach is that adequate day-to-day monitoring of the core power distribution can be done with ex-core instruments and restricted control rod configurations coupled with periodic (or monthly) use of moveable in-core detectors (thermocouples are also present at the outlet of selected fuel assemblies but the data appears to be infrequently used). The conversion of the detector signals to a power distribution is normally done off site. The number of assemblies that are accessible by the moveable in-core detectors (MIDS) represent about one quarter of the core. This system has the advantages of obtaining multiple axial points (24) and the capability to cross calibrate the various MIDS by traversing common fuel assemblies. A disadvantage is the long turn around time from performing the core map to receiving the inferred power distribution due to the offsite reduction of the data. This might impact plant availability in some instances where the power distribution must be obtained within a given time period, e.g., asymmetry caused by inserted rod. Also, a complete core map can take hours to complete and the results can be affected by drift from non-equilibrium core conditions and from the detectors themselves. This latter is sometimes alleviated by cross calibrating the MIDS two or three times during the map. The detectors are minature fission chambers and hence respond immediately to a change in flux, which allows the traverse to be performed at a reasonable speed. However, since they are the traversing type and normally out of the core, they are of little help tracking local power perturbations.

Babcock and Wilcox. The B&W approach is that the monitoring of the core power distribution can be done with a combination of in-core and ex-core instruments (thermocouples are sometimes provided at the outlet of selected fuel assemblies, however the data does not appear to be utilized very frequently). The in-core detectors are fixed at 7 axial locations in approximately 1/4 of the assemblies. The detector output is converted to a power distribution by a process computer on site. This system has the advantage of providing a power distribution (and all related quantities) upon demand and thus detailed information as to the core's status is readily available to the site engineers. It has the disadvantage of using only 7 axial points in determining the power distribution. The rhodium detectors have a response time which is on the order of minutes and hence cannot be used to monitor core transients or local perturbations which occur on a time scale smaller than the response time; i.e., 4 to 5 minutes. The B&W system uses rhodium detectors which in most of the current operating reactors are not cross calibratible and depend upon rigid manufacturing tolerances to assure the detector sensitivity (i.e., amps per nv) is a constant for all the detectors. In some B&W operating plants, there are indications that the sensitivity is not constant and this impacts the power distribution being inferred by indicating asymmetries that may not be real.

General Electric. The General Electric approach is to perform all monitoring functions using in-core instrumentation, both fixed and moveable. The fixed in-core detectors are positioned in four axial locations at the corner of four adjacent fuel assemblies. With this arrangement, approximately 25% of the fuel assemblies are adjacent to an instrument tube.

The detector output is converted to a power distribution by a process computer on line through the use of factors and algorithms which have been pre-determined off site. The fixed detectors are periodically cross calibrated by use of the moveable detectors. This system has the advantage of providing a power distribution and all dependent quantities on a frequent periodic basis and/or upon demand, which provides to the site engineer timely and detailed information as to the core's status. It has suffered, at times, from apparent mechanical displacement of the instrument tube which can lead to erroneous interpretation of the detector output and lead to power distributions that are not accurate. Also, the fixed detectors (which are fission detectors) tend to drift at times and unless noted by the reactor engineer and recalibrated can initiate a prohibit or block that can delay power rise. Being fission detectors, the response time is instantaneous and hence local neutron flux changes are monitored continuously.

Combustion Engineering. The Combustion Engineering approach is that the monitoring of the in-core power distribution can be done with a combination of in-core and ex-core instruments (thermocouples are provided at the outlet of selected fuel assemblies, but their output does not appear to be used on a routine basis). The in-core detectors are the self powered type (rhodium/vanadium) and are positioned at 4 axial locations in approximately 20% of the fuel assemblies. In some instances, C.E. also provides moveable in-core detectors which can access assemblies on the periphery of the core and aid in the verification of the axial power shape being inferred from the fixed detectors. The detector output is converted to a power distribution off site. In the latest plants, this is done via a data link between the site and the C.E. engineering offices. In this system the detector are calibrated before installation and there is no need for periodic calibration as long as depletion is accounted for properly. It has the disadvantage of using only 4 axial detectors which need to be converted to an axial power shape through analytical procedures. Also, even assuming quadrant symmetry, 50% of the assemblies are not "covered" by in-core instrumentation. Hence, the inferred power distribution is a result of more conversion factors and algorithms than the other systems.

As stated earlier, it is the authors opinion that this area of determining a detailed in-core power distribution and its subsequent utilization is becoming increasingly important to plant monitoring and will continue to affect operations in the future, impacting capacity factor and even availability.

The misconception that the in-core monitoring systems measure the core power distribution is shared by site operating engineers, design engineers and regulatory engineers, and the output from these systems is often accepted as absolute. While it is true that the output of these systems may be the best information available at the time, it is far from absolute or infallable, as has been addressed in the body of the report. As the dependence upon the in-core monitoring of the power distribution increases, problem areas, such as indicated deviations from design, will become major considerations. (This has already occurred in some operating plants.) In anticipation of this, there are specific areas that can be identified now, as being potential problem areas in the future.

- Detector Output - All the monitoring systems are based on the premise of a valid detector signal that is proportional to the neutron flux at a specific detector location. The evidence to date is that this premise is not always valid. There are indications that the detector is not always at the exact position assumed; that manufacturing tolerances are impacting output; that different type (i.e., fission and rhodium) detectors give different relative results at the same locations; that detector drift occurs which affects the relative output. These uncertainties get reflected in the end result, which is the inferred power distribution which is used to assure operating and safety limits which are being adhered to.

The cause or the extent and magnitude of these uncertainties does not appear to be well known even by the vendors; and whether some are common among vendors is not known. These uncertainties are being addressed by each of the vendors insofar as their particular system is concerned.

However, it seems prudent that the utilities themselves should also be concerned with this area, since it will have a direct effect on plant operations now and in the future; and that a systematic evaluation from series of plants over a period of time (at least one cycle) would provide valuable information that could be used to categorize these uncertainties.

- Detector Interpretation - As has been stated repeatedly, the parameters of interest to the reactor engineer and designer (the in-core power distribution and dependent quantities) are not directly measureable. What is sensed or measured is the reaction rate at the specific detector location. The detector current, which is proportional to the reaction rate, is converted to the power in the adjacent assembly, after being corrected for depletion, background, etc., by applying off-line calculated factors. The nodes or fuel segments not adjacent to the detector are inferred by symmetry considerations and/or extrapolation and interpolations from those that are. There are numerous assumptions and approximations in this conversion procedure which can affect the final result, the in-core power distribution. Again, since it is this power distribution that is governing the reactor operations from safety considerations to operating strategies, it would appear vital that the individual utility not only understand in detail this whole analytic procedure, but has available a sensitivity analyses for showing the impact of variations of these factors and coupling coefficients.

- Off-Line Analyses - The factors and coefficients that are used to interpret the detector signal and subsequently infer an in-core power distribution are generated off line by a complicated analytical procedure which is similar to that used in core design. Inherent in the generation of these factors are the assumptions and approximations that are part of any core analyses system. These include not only the analytical approximations to the geometry and neutron energy representation, but the judgment of the designer, such as how best to mock up the surrounding environment or what time detail is needed. As is evident these assumptions and judgments can eventually impact the inferred power distribution to whatever degree they effect the factors being generated.

The off-line analyses is the starting point for the eventual determination of an in-core power distribution. As mentioned in section 4.0, it can also be utilized as one of the means for aiding in the verification of the on-line results. In view of this and cognizant of the fact that eventually cores may have more than one vendors fuel, it would seem that the utility with an operating reactor should assure itself that access to this off-line capability is always readily available.

- Finally, as the body of the report discusses, quite different philosophies are inherent in the monitoring systems offered by each of the vendors. These include all in-core monitoring done on line, to most done off line, to in-core detectors of different types, to in-core and ex-core monitoring, to the eventual use of the final parameter, the in-core power distribution. Each of these systems has advantages and disadvantages and each has evolved semi-independently.

With the increases in importance of monitoring the in-core power distribution and a shifting of the requirements for utilizing the results, a fruitful area of investigation would be to address the question of is there an optimum system that might include parts of all the present systems and/or perhaps include different components than now exists.

6.0 REFERENCES

1. Peach Bottom 2 & 3 FSAR; Philadelphia Electric (version 10/75)
2. Hatch FSAR; Georgia Power
3. Process Computer Performance Evaluation Accuracy, J. F. Carew; NEDO -20340, June 1974
4. TROJAN FSAR, Portland General Electric
5. Prairie Island FSAR, Northern States Power
6. Evaluation of Hot Channel Factor Uncertainties, F. L. Langford and J. Bath, WCAP 7810, December 1971
7. Rancho Seco FSAR; Sacramento Municipal Power District (version 2/75)
8. St. Lucie FSAR; Florida Power and Light (version 8/75)
9. Private Communication FP&L - From Introduction of In-Core Instrument Analyses System
10. Operational Experience with Self Powered In-Core Nuclear Instrumentation, T. G. Ober, P. H. Gavin, ANS October 1973
11. On Line Power Distribution Monitoring with In-Core Detectors, J. R. Humphries, A. C. Kadak, R. W. Kwapp, ANS November 1975
12. St. LUCIE Technical Specifications; Florida Power and Light
13. Turkey Point FSAR, Florida Power and Light (version 6/75)

APPENDIX A

Examples of Technical Specifications Relating to Power Distribution

APPENDIX A

EXAMPLES OF TECHNICAL SPECIFICATIONS RELATING TO POWER DISTRIBUTIONS

The main body of the report addressed four general areas:

(1) The need for incore power distribution;

(2) Review of the current data acquisition systems;

(3) The acquisition or data requirements; i.e., reliability, accuracy impact of failure;

(4) The analytical requirements for converting the detector signals into a power distribution and subsequently other relevant parameters.

This was done in a qualitative manner (although Section 3 was more definitive) with the objective of providing the reader with a "feel" for the importance of information and the hardware and software problems associated with the various systems utilized by the four vendors for providing the information needed; i.e., the incore power distribution.

As Section 1.0 described, the in-core power distribution is the starting point for all the information needed for short, near and long term requirements. The short term requirements pertain specifically to the operational constraints which arise from safety and fuel integrity considerations. The technical specifications are the constraints based on safety considerations and are part of the license. Those governing the power distribution are complicated by the fact that they are a function of operating conditions; e.g., core power level, total coolant flow, etc. In addition, the action required when a limit is reached or exceeded (such as power reduction) also depends upon the operating conditions as well as the time the core has experienced the limit and the ability to respond with an updated inferred power distribution to verify the limiting condition is real. The Tech Specs for a B&W, C.E., G.E. or W type plant have different formats, different terminology and, in some cases, different requirements. It would be difficult to summarize these limits and consequences without the chance of misleading or misinforming the reader. To avoid this situation, the pertinent sections of the Tech Specs dealing with the power distribution and its derivatives for each of the four type reactors have been copied and are presented here in the same form as they appear in the license.

A.1 GENERAL ELECTRIC CONSTRAINTS

The information in this section is taken from the FSAR for Peach Bottom 2 & 3 (in Ref. 1).

The BWR's have a neutron flux scram setting and a rod block setting which is flow dependent and directly affected by the total peaking factor. There are limits on the APLHGR and local LHGR, the latter being axially dependent. These are directly impacted by uncertain or inaccurate power distribution determinations. There are also limits on the CPR which is dependent upon the bundle power as well as where the peak in the bundle occurs, both items coming directly from the inferred incore power distribution. The CPR has become extremely important in the higher power range because the limiting values have recently been changing from $\sim$ 1.26 to the 1.4 and 1.5 range, which is extremely difficult to maintain and has been restricting operations.

From Technical Specifications for a GE-BWR. (see Ref. 1).

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 <u>FUEL CLADDING INTEGRITY</u>	2.1 <u>FUEL CLADDING INTEGRITY</u>
<u>Applicability</u>	<u>Applicability</u>
The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.	The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.
<u>Objective</u>	<u>Objective</u>
The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding	The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.
<u>Specification</u>	<u>Specification</u>
	The limiting safety system settings shall be as specified below:
A. <u>Reactor Pressure $\leq$ 800 psia and Core Flow $\geq$ 10% of Rated</u>	A. <u>Neutron Flux Scram</u>
The existence of a minimum critical power ratio MCPR less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.	1. <u>APRM Flux Scram Trip Setting (Run Mode)</u> When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be: $S \leq 0.66\ W + 54\%$ where: S = Setting in percent of rated thermal power (3293 MWt) W = Loop recirculating flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr).
To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.	

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

A. Neutron Flux Scram (Continued)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.63, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \frac{2.63}{MTPF}$$

where:

MTPF = The value of the existing maximum total peaking factor

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

2. APRM - When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.

B. Core Thermal Power Limit (Reactor Pressure $\leq$800 psia)

When the reactor pressure is $\leq$800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

B. APRM Rod Block Trip Setting

$$S_{RB} \leq 0.66 + 42\%$$

where:

S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 x 10^6 lb/hr).

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	B. APRM Rod Block Trip Setting (Continued) In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.63, the setting shall be modified as follows: $S_{RB} \le (0.66\ W + 42\%)\ \frac{2.63}{MTPF}$ where: MTPF = The value of the existing maximum total peaking factor.
C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 inches above the top of the normal active fuel zone.	C. Scram and isolation - reactor low water level: $\geq$ 538 inches above vessel zero (0" on level instruments)
	D. Scram - turbine stop: $\leq$10 percent valve closure
	E. Scram - turbine control valve fast closure on loss of control oil pressure. $500 < P < 850$ psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.I. Average Planar LHGR	4.5.I. Average Planar LHGR
During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.5.1-A or 3.5.1-B, as applicable. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power. This daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.
3.5.J. Local LHGR	4.5.J. Local LHGR
During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power. This daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.
$LHGR \leq LHGR_d \, [1 - (\Delta P/P)_{max} (L/LT)]$	
$LHGR_d$ = Design LHGR = 18.5 kW/ft	
$(\Delta P/P)_{max}$ = Maximum power spiking penalty = <u>0.026</u>	
LT = Total core length = 12 ft Unit 2	
L = Axial position above bottom of core	

LIMITING CONDITION OF OPERATION	SURVEILLANCE REQUIREMENT

3.5.J. Local LHGR (Continued)

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

3.5.K. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be $\geq$1.26 at rated power and flow. For core flows other than rated the MCPR shall be $\geq$1.26 times k_f, where k_f is as shown in Figure 3.5.1-E. If at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding limits are again being met.

4.5.K Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power. This daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.

A.2 BABCOCK AND WILCOX CONSTRAINTS

The information in this section is taken from the FSAR for Rancho Seco (see Ref. 7)

The B&W-WPR reactor has limits on control configurations and power distributions and these are given in the following pages. There are limits on the quadrant tilt which are dependent upon operating conditions. There are also actions that are mandated when the tilt exceeds specific values. The frequency of monitoring the tilt is also specified. The axial imbalance monitoring frequency is specified along with the actions required when the limits are exceeded. Each of these items are obtained from the incore power distribution and are affected by any uncertainty or inaccuracies in determining the power distribution.

FROM TECHNICAL SPECIFICATION FOR A B&W - PWR (See Ref. 7)

LIMITING CONDITIONS FOR OPERATION

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown shall be not less than 1 percent $\Delta k/k$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

A. Operation with more than one inoperable rod as defined in Specification 4.7.1 and 4.7.2.3 in the safety or regulating rod banks shall not be permitted.

B. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of 1 percent $\Delta k/k$ hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first.

C. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1 percent $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

D. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised by a movement until indication is noted but not exceeding 2 inches within 24 hours and exercised weekly until the rod problem is solved.

E. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60 percent of the thermal power allowable for the reactor coolant pump combination.

F. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation above 60% of rated power may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.c.

3.5.2.3 The worth of a single inserted control rod shall not exceed 0.65 percent $\Delta k/k$ at rated power or 1.0 percent $\Delta k/k$ at hot zero power except for physics testing when the requirement of Specification 3.1.8 shall apply.

3.5.2.4 Quadrant tilt:

A. Whenever the quadrant power tilt exceeds 4 percent, escept for physics tests, the quadrant tilt shall be reduced to less than 4 percent within two hours or the following actions shall be taken:

(1) If four reactor coolant pumps are in operation, the allowable thermal power shall be reduced by 2 percent of maximum allowable power for each 1 percent tilt in excess of 94 percent maximum allowable power. Maximum allowable power is defined in Technical Specification.

(2) If less than four reactor coolant pumps are in operation, the allowable thermal power shall be reduced by 2 percent of maximem allowable power for each 1 percent tilt below the power allowable for the reactor coolant pump combination.

(3) Except as provided in 3.5.2.4.b, the reactor shall be brought to the hot shutdown condition within four hours if the quadrant tilt is not reduced to less than 4 percent after 24 hours.

(4) The power range high flux set point shall be reduced 2 percent of the maximum allowable flux for the RC pump combination for each 1 percent tilt in excess of 4 percent.

B. If the quadrant tilt exceeds 4 percent and there is simultaneous indication of a misaligned control rod per Specification 3.5.2.2, reactor operation may continue, provided power is reduced to 60 percent of the thermal power allowable for the reactor coolant pump combination.

C. Except for physics tests, if quadrant tilt exceeds 9 percent, the reactor shall be brought to the hot shutdown condition within four hours.

D. Whenever the reactor is brought to hot shutdown pursuant to 3.5.2.4a(3) or 3.5.2.4c above, subsequent reactor operation is permitted for the purpose of measurement, testing and corrective action provided the thermal power and the power range high flux setpoint allowable for the reactor coolant pump combination are restricted by a reduction of 2 percent of maximum allowable power for each 1 percent tilt.

E. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15 percent of rated power.

3.5.2.5 Control Rod Positions

A. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

B. Operating rod group overlap shall be 25 percent +5 percent between two sequential groups, except for physics tests.

C. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on figure 3.5.2-1. If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall then be attained within two hours.

D. Except for physics tests, power shall not be increased above the power level cut-off of 94 percent of the maximum allowable power level unless one of the following conditions is satisfied:

(1) Xenon reactivity is within 10 percent of the equilibrium value for operation at the maximum allowable power level and asymptotically approaching stability.

(2) Except for xenon-free startup, when 3.5.2.5d(1) applies, the reactor has operated within a range of 89 to 94 percent of the maximum allowable power for a period exceeding 2 hours in the soluble poison control mode.

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by figure 3.5.2-2. If the imbalance is not within the envelope defined by figure 3.5.2-2, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are set.

A.3 COMBUSTION ENGINEERING CONSTRAINTS

The information in this section is taken from the FSAR for St. Lucie (see Ref. 8, 12).

The C-E PWR plant has a limit on the linear heat rate and specifies the actions to be taken when it is exceeded. The surveillance requirements are specified for both incore and excore systems.

There are limits on the total radial peaking factor and the actions required are specified along with the surveillance requirements. The "measured" peaking factor enters directly into the thermal power allowed and the setpoints on the power ratio trip.

The Aximuthal Power Tilt limit is specified along with the actions required and the surveillance requirements. The "measured" total radial peaking factor appears in the set point settings and since the tilt is part of F_r^T, inaccurancies impact directly on this limit.

FROM TECHNICAL SPECIFICATION FOR A C-E PWR (See Ref. 8, 12)

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent limits on the Power Ratio Recorder, immediately initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or
b. Be in HOT STANDBY within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.2 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.
b. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the determination made in Specification 4.2.1.2.c, and
c. Verifying at least once per 31 days that the THERMAL POWER does not exceed the value determined by the following relationship:

$$\frac{L}{17.0} \times M$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

2. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

4.2.1.3 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

SURVEILLANCE REQUIREMENTS (Continued)

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

   1. Flux peaking augmentation factors as shown in Figure 4.2-1,

   2. A measurement-calculational uncertainty factor of 1.08,

   3. An engineering uncertainty factor of 1.03,

   4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

   5. A THERMAL POWER measurement uncertainty factor of 1.02.

TOTAL RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_r^T, defined as $F_r^T = F_r^P(1+T_q)$, shall be limited to $\leq$ 1.36.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With F_r^T > 1.36, within 4 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3, fully withdrawn the PLCEAs and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Reduce the Local Power Density - High and Thermal Margin/Low Pressure trip setpoints and the setpoints on the Power Ratio Calculator by a factor equivalent to $\geq \dfrac{F_r^T \text{ (meas)}}{1.36}$, or

c. Be in HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_r^T shall be calculated by the expression $F_r^T = F_r^P(1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days, and

c. Within four hours of the AXIMUTHAL POWER TILT (T_q) is $\geq$ 0.02.

4.2.2.2 F_r^P shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with no part length CEAs inserted and with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.2.3 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

*See Special Test Exception 3.10.2

POWER DISTRIBUTION LIMITS

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.02.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.02 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL RADIAL PEAKING FACTOR (F_r^T) is within the limit of Specification 3.2.2.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided F_r^T or the combination of F_r^T and THERMAL POWER is maintained within the limit of Specification 3.2.2. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided:

   1. The THERMAL POWER level is restricted to < 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination, and

   2. The Local Power Density-High and Thermal Margin/Low Pressure trip setpoints and Power Ratio Calculator setpoints are reduced by a factor equivalent to $\geq \dfrac{F_r^T \text{ (meas)}}{1.36}$.

SURVEILLANCE REQUIREMENT

4.2.3.1 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days when the Subchannel Deviation Alarm is OPERABLE,

b. Calculating the tilt at least once per 12 hours when the Subchannel Deviation Alarm is inoperable, and

c. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

*See Special Test Exception 3.10.2

A.4 WESTINGHOUSE CONSTRAINTS

The information in this section is taken from the FSAR for Turkey Point (see Ref. 13).

The W-PWR has limits on the hot channel factors which are functions of core power level and the axial location of Fq. In addition the actions required are specified, including how soon an incore map must be made. There are limits on the axial flux difference which are power level dependent and the actions required are also specified. The limit on the quadrant tilt is defined and the actions required when exceeded are specified. Each of these limits is obtained directly from the incore power distribution and are affected by inaccurancies in the "measured" values. In Westinghouse's case, the uncertainty is somewhat greater because all day-to-day monitoring of these limits is done by excore instrumentation which is periodically (monthly) checked via an incore map.

FROM TECHNICAL SPECIFICATION FOR A W-PWR (See Ref. 13)

6. POWER DISTRIBUTION LIMITS

a. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$Fq(Z) \leq (2.32/P) \times K(Z)$ for $P > .5$

$Fq(Z) \leq (4.64) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$

where P is the fraction of design power at which the core is operating. K(Z) is the function given in Figure 3.2-3 and Z is the core height location of Fq.

b. Following initial loading before the reactor is operated above 75% of rated power and at regular effective full rated power monthly intervals thereafter, power distribution maps, using the moveable detector system shall be made, to conform that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

(1) The measurement of total peaking factor, F_q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

(2) The measurement of the enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error.

If either measured hot channel factor exceeds its limit specified under Item 6a, the reactor power shall be reduced so as not to exceed a fraction of the rated value equal to the ratio of the F_q or $F^N_{\Delta H}$ limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent incore mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing. The reactor may be returned to higher power levels when measurements indicate that hot channel factors are within limits.

c. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation using the most recent measured value and the value predicted for the end of the cycle life.

d. Except during physics tests or during excore calibration procedures and as modified by items 6e through 6g below, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference (this defines the target band on axial flux difference).

e. If the indicated axial flux difference at a power level greater than 90% of rated power deviates from its target band, the flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

f. At a power level no greater than 90% and above 50% of rated power,

   1. The indicated axial flux difference may deviate from its ± 5% target band for a maximum of sixty effective minutes (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11% and +11% at 90% of rated power and increasing by -1% and +1% for each 2% of rated power below 90% rated power.

      Effective time out of the target band is defined as the sum of the time out of the target band at power levels above 50% plus one half the time out of the target band at power levels of 50% and below.

   2. If item 1 above is violated, then the reactor power shall be reduced to no greater than 50% of rated power and the high neutron flux setpoint reduced to no greater than 55% of rated power.

   3. A power increase to a level greater than 90% of rated power is contingent upon the indicated axial flux difference being within its target band.

g. At a power level no greater than 50% of rated power,

   1. The indicated axial flux difference may deviate from its target band.

   2. A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than sixty effective minutes (cumulative) out of the preceding 24 hour period..

h. If the quadrant to average power tilt exceeds a value of 2% except for physics and rod exercise testing, then:

   1. The hot channel factors shall be determined within 2 hours and the power level and trips adjusted to meet the requirements of Item 6a.

2. If the hot channel factors are not determined within two hours, the power shall be reduced from rated power 2% for each percent of quadrant tilt.

3. If the quadrant to average power tilt exceeds ± 10%, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

i. If after a further period of 24 hours, the power tilt in k. above is not corrected to less than +2%, and

1. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Nuclear Regulatory Commission.

2. If the hot channel factors are not determined, the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

7. INCORE INSTRUMENTATION

a. A minimum of 16 thimbles, at least 2 per quadrant, and the necessary associated detectors shall be operable during the check and calibration of nuclear instrumentation ion chambers.

b. Power shall be limited to 90% of rated power for 3 loop or 50% of rated power for 2 loop operation if the requirements in Section 7.a are not met.

APPENDIX B

SUPPLEMENTARY QUESTIONS: ON-LINE CORE MONITORING

(RESULTS ENGINEER AT NUCLEAR PLANTS)

I. POWER DISTRIBUTION INSTRUMENTATION:

A. EXCORES: type ___________, #/string ___________, # of strings ___________.

B. INCORES:

1. fixed: type __________, #/string __________, # of strings __________.

2. moveable: type __________, # _____, total # of channels which may be traversed __________.

3. temperature: type __________, # __________, general locations ____________________________________

4. flow: type __________, # __________, general locations ____________________________________

5. (a) Do you have moveable incores? Yes ☐ No ☐

   (b) Can they be intercalibrated in a common channel? Yes ☐ No ☐

   (c) Can they be used to calibrate the fixed detectors? Yes ☐ No ☐

   (d) Are there fixed detectors? Yes ☐ No ☐

C. RELIABILITY:

1. Type of detector: How many failures have you had in 1975?

Type of detector	How many failures have you had in 1975?

2. If you have moveable detectors describe problems, if any (MTBF, MTTR, etc.).

3. Describe your feelings about the reliability of incore and excore instrumentation.

4. (a) Have detector failures caused you problems ranging from nuisance to operating restrictions? Yes ☐ No ☐

   (b) Please describe:

II. CORE MONITORING AND PERFORMANCE CALCULATIONS:

A. TYPE OF CALCULATION	FREQUENCY							ADEQUACY					REQUIRED					PERFORMED ON		
(Please check appropriate columns)	Continuous	Hourly	Daily	Weekly	Monthly	Never Used	Don't Have	Good	Adequate	Poor	Not Acceptable	Wish We Had	By Tech. Spec.	For Start-up	For Power Change	For Operating History	Optional	Process Computer	Auxiliary On-Line Computer	Off-Line Computer
1. Peak Linear Heat Rates																				
2. DNBR Margin																				
3. Fuel Pre-conditioning																				
4. Axial Off-set																				
5. Fuel Burnup																				
6. Detailed Heat Balance																				
7. Detailed Flux Mapping																				
8. Inter Detector Calculation																				
9. Control Rod Worth																				
10. Boration Level Control																				
11. Other:																				

B. To what extent is plant operation dependent upon the calculations you have identified above as being performed on-line?

____

____

____

C. If the process computer is not available, what are the implications? __________

____

____

____

D. Rate the adequacy of the following:
(Please check appropriate column)

	Good	Adequate	Poor	Needs Improvement	Unimportant
1. Detector reliability.					
2. Detector accuracy and consistency.					
3. Capability to maintain required azimuthal flux symmetry.					
4. Time required for on-line calculations.					
5. Ability to obtain required calc. parameters.					
6. Capability to store and transfer data to off-line system.					
7. Adequacy of information presentation by on-line system.					
8. Availability of on-line calculations.					
9. Degree to which calculations satisfy your needs.					
10. Other.					

E. Please describe the <u>three</u> most successful attributes of the core monitoring and performance system:

(1) ______________________________

(2) ______________________________

(3) ______________________________

F. Have you had any serious problems with the core monitoring and performance system? Yes ☐ No ☐

If so, please describe the three most severe:

(1) ______________________________

____

(2) ______________________________

____

(3) ______________________________

____

G. Would you change or implement this system differently? Yes ☐ No ☐

If so, how? ______________________________

____

____