WESTINGHOUSE CLASS 3

WCAP-11847

RCS FLOW ANOMALY
INVESTIGATION REPORT
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1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this report is to document the investigation of an anomalous condition observed in several Westinghouse 4-loop plants. This condition has been observed in Nuclear Instrumentation System (NIS) data and in measured parameters in the Reactor Coolant System (RCS). This report describes the investigations undertaken, the data obtained from operating plants, the results of the data analysis, the safety evaluations, and the overall conclusions of the studies.

1.2 BACKGROUND

In late November, 1986, with the plant at approximately 80% of full power, personnel at Union Electric's Callaway plant were investigating the occurrence of flux deviation alarms. Strip chart recordings of the excore flux data showed distinct step changes occuring randomly in one or more quadrants. Union Electric personnel subsequently collected selected plant information from excore detectors, incore detectors, core exit thermocouples, loop flows, loop temperatures, Reactor Vessel Level Instrumentation System (RVLIS) pressure differentials, Reactor Coolant Pump (RCP) current, and the loose parts monitoring system.

Review of the data revealed simultaneous changes in several parameters, occuring randomly every 2 to 5 minutes. A summary of the observations is described below:

b,g

Excore detectors
Incore detectors

Core exit T/Cs
RCS flow
RVLIS Train A
RVLIS Train B

Figure 1-1 shows a typical strip chart of the data observed for what was termed an N-44 event; N-44 indicating the quadrant with the largest excore flux decrease. As noted, a decrease in the N44 channel excore flux detector is accompanied by a simultaneous decrease in the RCS flow and an increase in the L-12 core exit thermocouple which is in the quadrant adjacent to the N44 detector. Also an increase is observed in the RVLIS A channel and a decrease in the RVLIS B channel. RVLIS Train A lower tap is located in the lower vessel head near the center of the vessel, while the Train B lower tap is located in the lower head near the center of the core quadrant adjacent to excore detector N44. Figure 1-2 shows similar data on a core map. Typically, the significant changes occur simultaneously in a particular core quadrant. No indications were observed in the loose parts monitoring system. No variations in RCP motor currents, pump speed, or instrument power supplies were observed.

Westinghouse was contacted by Union Electric(UE) on November 26, 1986 to provide support in the investigation of the anomalous behavior and to support continued operation of the Callaway plant. Westinghouse dispatched a three man team to Callaway to record additional data and make recommendations for continued monitoring. In addition Westinghouse performed a safety evaluation to support Union Electric's justification for continued operation (JCO). As a result of executive level discussions between Union Electric and Westinghouse, a Westinghouse task team was formed to continue the investigations into the Callaway flow anomaly and to work with Union Electric to determine the cause of the anomaly.

The task team made recommendations to Union Electric regarding measurements to be taken during power increases or reductions, data collection, data analysis, and continued monitoring. Westinghouse worked with UE personnel from the St. Louis office and the Callaway site to establish a matrix of potential causes of the flow anomaly and with continued investigations and additional data, to eliminate the potential causes not supported by the data. Members of the task team were also on site at Callaway to obtain more data, particularly neutron noise measurements. A report was provided to UE on the results of the neutron noise data analysis.

The data analysis is described in detail in Section 3 of this report. The data utilized by the investigation team consisted of the following:

RCS Parameters (flows, temperatures)
Core exit thermocouple data
NIS data (excore and incore detectors)
RVLIS data
Loose parts monitoring data
Low frequency nuclear noise data
Nuclear noise data

By systematic investigation of the data (including data from the Wolf Creek plant), the following potential causes were eliminated:

Instrumentation bias
Mechanical blockage in the primary system
Foreign objects/loose parts
RCP performance reduction
Fuel assembly or lower internals motion
Secondary side perturbations
Neutronics anomalies (e.g. dropped rodlets)
Voiding in the core
Flow changes in RCS due to charging system or pressurizer

The probable cause of the anomalous behavior was concluded to be an aperiodically occurring vortex type flow disturbance in the lower plenum. The observed changes in the RCS and NIS parameters are due to the appearance and disappearance of the vortex in the lower plenum under a particular core quadrant. The flow disturbance resulted in a reduction in inlet flow to a number of fuel assemblies. The effect of the flow disturbance is to cause a slight reduction in DNB margin, which must be considered in the safety analysis, and a change in the velocity field in the lower plenum, which must be considered in the structural analysis of the lower internals. Details of these evaluations are provided in Section 4.

1.3 NRC ACTIVITIES AND ADDITIONAL PLANT DATA COLLECTION

Union Electric contacted the NRC upon first discovering the existence of the flow anomaly. The Staff requested Union Electric to provide a Justification for Continued Operation (JCO) for Callaway. In addition, the NRC requested Union Electric to submit a description of the conditions under which Union Electric would shut down Callaway in the event the anomaly worsened. This information was submitted to the NRC by UE with support from Westinghouse.

The NRC also expressed interest as to whether the anomaly was present at other plants. Wolf Creek, which was in an outage at the time, was discussed as a possible candidate for additional data collection since it is a sister unit to Callaway. Once Wolf Creek restarted, data was taken and it was determined that it, too, displayed anomalous behavior in the RCS. Duke Power and Northeast Utilities were then contacted and requested to take data. Data was also collected at Commonwealth Edison's Byron Unit 1. The data indicated that the flow anomaly was present at Catawba 1 and 2, but not at the Byron 1, Millstone 3 or the McGuire units.

On January 16, 1987, a meeting was held with the NRC, Union Electric, Duke Power, Northeast Utilities, Wolf Creek Nuclear Operating Corporation, and Westinghouse. Plant data, the Callaway safety evaluation, suspected reason for of the anomaly, and future plans were discussed at this meeting. The Westinghouse slides from this meeting are documented in Reference 1. Westinghouse described the elements of a plan to investigate the anomaly. UE requested relief from their monitoring requirements and from their "action levels" for reducing power at Callaway. The Staff was reluctant to permit relief to Union Electric at the meeting, however, UE was notified within approximately two weeks that they did not have to observe the action levels and could reduce monitoring for the anomaly. The Staff requested Westinghouse to recommend other plants for data collection. A request was made to nine other plants (the Group 2 plants) shown in Table 1-1. In this group, only Indian Point Unit 2 showed anomaly type indications and these with substantially reduced magnitude. In an unrelated activity, RCS flow data, collected by Houston Lighting and Power (HL&P) during hot functional testing at South Texas Unit 1, showed fluctuations similiar to those produced by the anomaly, suggesting that additional data be taken during cycle 1 operation.

Westinghouse communicated information on the flow anomaly to all utilities with Westinghouse plants through the Westinghouse Owner's Group (WOG). Communiques were transmitted to the WOG on January 12, 1987 and February 17, 1987 on the NETWORK system. In addition presentations were made to the WOG in January and the WOG Analysis Subcommittee in March of 1987.

1.4 FORMATION OF TASK TEAM

On December 19, 1986 Westinghouse formed a Task Team to investigate the generic implications of the flow anomaly. The Task Team's objective was to determine the plants affected and the most probable cause. The remaining sections of this report present the detailed results of the Task Team efforts.

TABLE 1-1
PLANT DATA

Plant	Anomaly Observed	Group	Comment
Callaway	Yes	1	4 loop, flat LSP*
Wolf Creek	Yes	1	4 loop, flat LSP
Catawba 1	Yes	1	4 loop, flat LSP
Catawba 2	Yes	1	4 loop, flat LSP
McGuire 1	No	1	4 loop, flat LSP
McGuire 2	No	1	4 loop, flat LSP
Millstone 3	No	1	4 loop, flat LSP
Byron 1	No	1	4 loop, flat LSP
Indian Point 2	**	2	4 loop, dome LSP, Cycle 8
Diablo Canyon 1	No	2	4 loop, dome LSP
Diablo Canyon 2	No	2	4 loop, flat LSP
Indian Point 3	No	2	4 loop, dome LSP (asymmetric Steam Generator Tube Plugging)
Trojan	No	2	4 loop, flat LSP, Cycle 8
Shearon Harris	No	2	3 loop, flat LSP
Beaver Valley	No	2	3 loop, flat LSP, Cycle 6
Prairie Island	No	2	2 loop, Cycle 10
Ginna	No	2	2 loop, Cycle 10
Byron 2	No	3	4 loop, flat LSP
South Texas	***	3	4XL, flat LSP (hot functional test)

* Lower Support Plate

** Condition probable at substantially reduced magnitude.

*** Indications observed, additional data needed for confirmation.

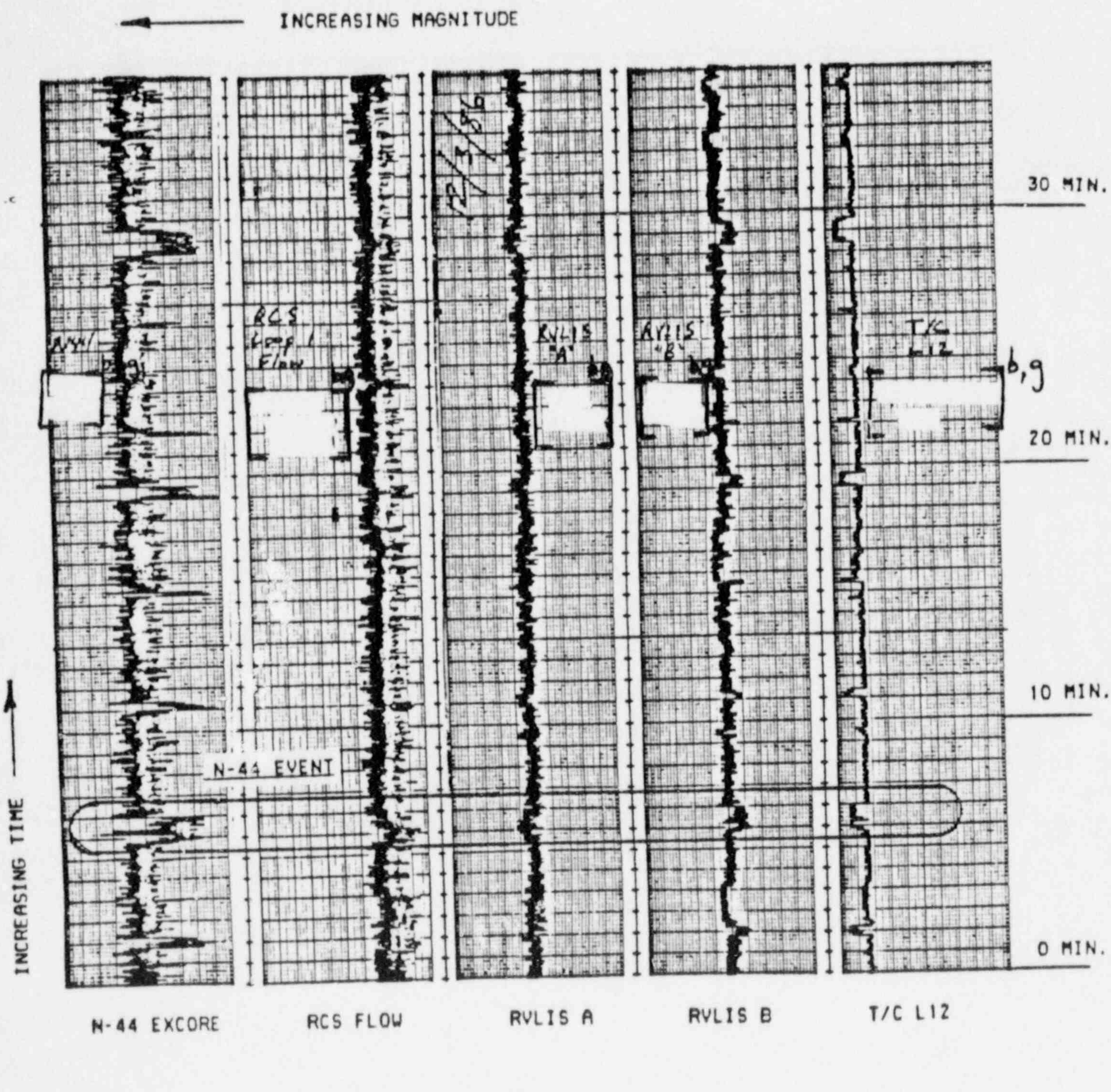

Figure 1-1 Typical Strip Chart of Data

b,g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 1-2 Typical Core Map of Data

2.0 SUMMARY AND CONCLUSIONS

The investigation of the RCS Flow Anomaly incorporated information and data collected during the evaluation of the events observed at Union Electric's Callaway Unit. This included the results of the nuclear noise evaluation, the conclusions from the causal matrix activity, the core x-y correlation of plant measurements, and the preliminary screening of related sub-scale hydraulic test data. The activities of the investigation were directed at determining the type of plants involved, evaluating the DNB penalty with a mechanistic core inlet flow distribution, and developing a more complete understanding of the established probable cause, i.e. a flow disturbance due to a rotational cell or cells in the reactor lower plenum.

A second group of plants was identified at which a select list of data was requested to establish the extent of the issue. In this list 2-loop, 3-loop, and the two types of 12 foot core 4-loop reactor vessel internals were included. Subsequently, hot functional testing at Houston Light & Powers' South Texas Unit provided data on the 14 foot core style 4-loop plants. Additional data was also collected at Duke Powers' Catawba Units 1 and 2 to fully characterize the core outlet temperature effects and neutron detector outputs in plants with nearly constant rotational flows.

An analysis of the Callaway core exit thermocouple data and incore flux data was performed with the THINC Code. Initially a uniform reduction of core inlet flow over a group of assemblies was modeled. Later, a variable reduction in core inlet flow was used. The variable core inlet flow distribution was obtained from a correlation of sub-scale core inlet flow test data in which vortex flows were observed. Also included were cycle specific core power distributions from Callaway.

In order to fully describe the probable cause, data from plants where the events could be clearly identified, was reduced and summarized. This data was combined with information from sub-scale hydraulic tests and other relevant hydraulic information. The results confirmed that the flow disturbance is due to multiple rotational flow cells (vortices) in the reactor lower plenum. In 4-loop plants with flat core support plates, the development of these cells can be attributed to a combination of several factors: 1) the pairing of the inlet nozzles producing a local peak in downcomer flow velocity, 2) the position of the neutron pads and radial keys which provide a preferred flow path to the lower plenum, 3) a non-symmetric tie plate design in the lower plenum which makes the N-44 and N-42 quadrants preferred locations for the rotational flows*, 4) a reduced lower plenum structural density (thinner tie plates and BMI columns) which reduces flow energy dissipation rates, and 5) plant specific loop to loop flow imbalances which modulate the tie plate and column effects and cause plant to plant variations in the observed phenomena. In some 4-loop plants with the dome core support plates, the paired inlet nozzles, radial key positions, and loop to loop flow imbalances appear sufficient to produce similar phenomena but with more slowly varying characteristics and substantially smaller magnitude than in the flat core support plate plants.

As a result of this investigation, the following conclusions have been developed:

1. The unsteady flow anomaly as observed at Callaway, Wolf Creek and the Catawba Units is caused by a rotational flow cell or cells in the reactor lower plenum which cause local disturbances in core inlet flow and lead to the observed data fluctuations.

2. Investigation of the Callaway and Wolf Creek data, including measurements of neutron noise, eliminated: instrumentation bias, foreign objects or loose parts, motion of reactor internals or fuel, secondary side perturbations, RC pump performance changes, and neutronics anomalies as possible causes of the flow disturbance.

3. The unsteady RCS flow anomaly is limited to a specific group of 4-loop plants. The reactor internals geometry in all 2-loop and all domestic 3-loop plants do not contain the paired vessel inlet nozzles and the low

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* As indicated in Table 3-3(A), columns 3, 4, and 5.

structural density lower internals design, i.e. thinner tie plates and BMI columns, both significant contributors to the 4-loop condition. Data collected from two 2-loop plants and two 3-loop plants has confirmed the absence of the anomaly.

4. Significant forms of the RCS flow anomaly appear limited to the flat core support style 4-loop plants. Of the three sets of data collected at dome core support style 4-loop plants, only one showed any indications of anomaly type events. In that case, the anomaly had no effect on loop flow and the events, where discernable, were of a substantially reduced magnitude and did not have the step like character of events in the flat core support style plants.

5. The RCS flow anomaly has not been observed in 4-loop plants with a high structural density of lower plenum hardware, i.e. plants with 3" thick tie plates and 3.75" diameter Bottom Mounted Instrumentation (BMI) columns.

6. The fraction of time with the dominant event condition (N-44/42)* varies from 0% to nearly 100% in plants with reduced lower plenum structural density. Loop to loop flow imbalances tend to suppress vortex formation in these plants. The steady and unsteady anomalies appear to be limited to plants with less than approximately [ ]^{b,g} loop to loop flow imbalance. N-1 loop operation also tends to suppress the occurance of the anomaly.

7. Based upon an evaluation of all data from plants where the anomaly was observed, the core inlet flow maldistribution results in a maximum DNB penalty of [ ]^{c} Accounting for [ ]^{c}

* N-44/42 indicates an event with a pair of vortices, the principal vortex listed first.

[

]^c typically. Plant specific evaluations will be required to determine the applicable loss in DNB margin. However, [

]^c

8. The flow anomaly has no impact on large and small break LOCA analysis.

9. The structural integrity of the reactor internals is not expected to be affected by the flow anomaly since it appears sufficient fatigue margin exists and the cross flow effects on the fuel rods are within acceptable limits.

10. A diagnostic technique for determining the presence of steady rotational flows, based on low frequency nuclear noise analysis, provides a mechanism for evaluating units which exhibit no unsteady conditions.

3.0 ANALYSIS OF DATA

3.1 EVALUATION MATRIX

During the initial investigation of the reactor coolant parameter anomalies at the Callaway plant, several postulation mechanisms for the anomalies were identified jointly by Union Electric and Westinghouse personnel. The mechanisms possible were evaluated to determine if they were consistent with the behavior of key coolant parameters observed at the plant, described below:

Concurrent reductions in all reactor coolant loop flows.

Concurrent decrease in excore power, movable detector signals, and increase in core exit temperatures in the same quadrant of the core.

Anomaly appears in different quadrants, and sometimes shifts from one quadrant to another during an event.

RVLIS pressure decreased under the core when the anomaly was in the quadrant with the RVLIS connection; pressure increased when the anomaly was in a different quadrant.

For most of the possible mechanisms, the plant data provided the necessary information to clearly indicate that there was no correlation, and these postulated mechanisms were discounted. In some cases, data from the Wolf Creek plant, where anomalies had also been detected, provided additional support for the evaluations. As the evaluations progressed and most of the postulated mechanisms were eliminated, the effort could more quickly evaluate the more likely mechanisms. The evaluations eventually led to the elimination of all but one: a flow disturbance in the lower plenum of the reactor vessel, producing local reductions in flow entering the core. Additional data and information subsequently provided added confirmation of this phenomena as the basis of the observed anomalies.

The list of the postulated mechanisms and reasons for eliminating all but the single cause is summarized in Table 3-1.

TABLE 3-1
EVALUATION MATRIX

1. FLOW BLOCKAGE: Flow into some fuel assemblies partially blocked due to a foreign object in the downcomer or lower plenum, or a displacement of a neutron pad or surveillance specimen.

   This was discounted for the following reasons:

   a. The anomaly changes location, so a foreign object would have to change location, however there were no indications from loose parts monitors.
   b. Anomalies were aperiodic, whereas flow would tend to hold a loose part stationary.
   c. Top and bottom excore detector current ratios from prior and current fuel cycle do not indicate a displaced neutron pad.
   d. Surveillance specimen was removed during the Cycle 1-2 refueling without difficulty.
   e. Other plants are experiencing similar anomalies.

2. VARIATIONS IN CORE BYPASS FLOW: Anomaly results from variations in barrel-baffle bypass, upper head bypass or hot leg nozzle bypass flow.

   This was discounted for the following reasons:

   a. Comparison of core vs. loop delta Ts does not indicate correlation with bypass flow during events.
   b. Not consistent with observed decrease in RCS loop flows.
   c. Cannot explain anomaly movement between core quadrants.
   d. Barrel-baffle bypass flow variations would affect fuel assemblies nearest the baffle. This was inconsistent with incore flux and core exit T/C data.

3. FLOW VARIATION DRIVEN BY RC PUMPS:

   This was discounted for the following reasons:

   a. No change observed in pump currents or pump speed.
   b. Cannot explain anomaly movement between core quadrants.
   c. Data from other plants tend to rule this out.

TABLE 3-1 (continued)
EVALUATION MATRIX

4. FREQUENCY/VOLTAGE CHANGES ON INSTRUMENT BUSES:

This was discounted for the following reasons:

a. Power supplies show no deviation.
b. Some instruments indicate increases, some decreases.
c. Data from other plants tends to rule this out.
d. Anomaly observed on separate IE channels and on non-IE channels with separate power sources.

5. CHARGING FLOW/TEMPERATURE FLUCTUATIONS:

This was discounted for the following reasons:

a. Regen HX outlet temperature indicates no fluctuations.
b. No observable charging flow changes correlate with events.
c. Does not explain variations in all four RCS loop flows.

6. PRESSURIZER SURGE LINE FLUCTUATIONS:

This was discounted for the following reasons:

a. Not supported by pressurizer pressure or heater response.
b. Does not explain variations in all four RCS loop flows.

7. OFA/STD FUEL ASSEMBLY CROSSFLOW:

This was discounted for the following reasons:

a. Excore detector data from the end of Cycle 1 indicate that the same anomaly existed, prior to loading Optimized Fuel Assemblies (OFA).
b. Wolf Creek has no OFA, yet has a similar anomaly.

TABLE 3-1 (continued)

EVALUATION MATRIX

8. EXCESSIVE CORE BARREL AND LOWER INTERNALS MOTION:

This was discounted for the following reasons:

a. Neutron noise analyses show consistent behavior with another 4-loop plant which has no observed anomaly.
b. No mechanism postulated to correlate motions with anomaly.

9. CORE SUPPORT STRUCTURE MOVEMENT:

This was discounted for the following reasons:

a. No mechanism postulated to correlate motions with anomaly.
b. Absence of loose parts monitor indications.
c. Not supported by neutron noise data.

10. FUEL ASSEMBLY HOLDDOWN SPRING/MOTION:

This was discounted for the following reasons:

a. Cannot explain total RCS flow change.
b. Inconsistent with RVLIS, loop flow and core exit temperature indications, and with anomaly movement.

11. SECONDARY SIDE PERTURBATIONS:

This was discounted for the following reasons:

a. Secondary parameter changes would not affect RCS flows.
b. Observed changes in coolant cold leg temperature are not of sufficient magnitude to generate an anomaly.

TABLE 3-1 (continued)
CAUSAL MATRIX

12. DROPPED RCCA RODLETS:

This was discounted for the following reasons:

a. No indications on flux maps.
b. Does not explain RCS flow or RVLIS indications.
c. Does not explain anomaly movement between quadrants.
d. Data from other plants tends to rule this out.
e. Anomaly existed during Cycle 1, with no dropped rodlets.

13. VOIDING IN THE CORE:

This was discounted for the following reasons:

a. RCS flow changes at 30% power are the same at 100% power.
b. RCS flow changes present at zero power at Wolf Creek.

14. FLOW DISTURBANCE IN REACTOR VESSEL:

This mechanism is believed to be probable since:

a. Evaluation of low frequency (1/2 Hz range) neutron noise data and RVLIS indications supports aperiodic hydraulic instability in the reactor vessel lower plenum.
b. Explains occurrence of anomaly at other plants.
c. Provides only correlation with all observed indications.

3.2 PROBABLE FLOW ANOMALY MECHANISM

3.2.1 Introduction

As part of the RCS Flow Anomaly investigation, data from Callaway and Wolf Creek was reduced and summarized in an effort to identify the probable mechanism. This data, with information from subscale hydraulic tests and other relevant hydraulic information, strongly suggested that the flow disturbance emenates from one or more rotational flow cells (vortices) in the reactor lower plenum which are influencing the core inlet flow. In order to fully evaluate the rotational flow cells, a comprehensive model of the observed phenomena and initiating mechanisms was developed. Since the anomaly has not been observed in 2 and 3-loop plants, it was based upon a review of relevant 4-loop plant data combined with information from 4-loop sub-scale hydraulic tests. The result, which has been labeled an "Empirical Model of the Flow Anomaly", provides a mechanistic explanation of the observed events at Callaway, Wolf Creek, and the Catawba Units 1 and 2. While the model includes a number of extrapolations from tests of non-prototypic hardware and applications of data with substantial uncertainty bands, the result is a highly probable explanation of the anomaly.

In this evaluation, event types are designated by the Excore Detector numbers for the quadrants in which vortices are present, with the principal vortex listed first. The nomenclature denc s the existance of the secondary (and apparently weaker vortices) that were not included in the original identification system. Thus, the N-44/42 is the same condition as the original N-44 and similiarly with the N-43/41 which was the N-43.

3.2.2 Probable Mechanism

Intermittent rotational flow cell (vortex) formation and breakup is considered to be the mechanism for the Hydraulic Flow Anomaly based on the following considerations:

1. Comparisons of the Callaway core exit T/C data, RVLIS differential pressures, incore movable detectors, and excore detectors, shows a strong correlation with vortex characteristics for the significant event types; N-44/42, N-43/41, and N-44/43/42.

2. The random periodicity observed at Callaway, Wolf Creek, and the Catawba Units is similar in characteristic to the random vortex periodicity previously observed in a non-PWR test facility. The ramp like buildup and breakdown (over 1 to 3 sec) of the event are also similar to loop flow behavior previously observed.

3. Downcomer and lower plenum flow distribution tests of 4-loop style reactors have shown fluctuating vortices which impacted the core inlet flow distribution under specific conditions. When the BMI columns and/or tie plates were removed creating a plenum free of hardware, rotational cells of increased vorticity have been observed.

4. THINC analysis of the Callaway core which simulates the observed nuclear and hydraulic phenomena indicates that a significant core inlet flow reduction exists in the plant. This type of flow reduction has been associated with vortices observed in subscale hydraulic tests of other reactor designs.

5. The lack of Loose Parts Monitor indications suggests that a physical blockage is not present.

3.2.3 Data Evaluation

A compilation of the 4-loop plant data, collected since the initial observations at Callaway, has been tabulated in Tables 3-2 and 3-3. In Table 3-2, the types of data obtained are listed and the events in representative sets of plant data are described. In Table 3-3, a quantitative evaluation of these events is provided in which the following items are listed:

1. Operating Condition - Steady event, unsteady event, or no event.
2. Event Type - Quadrants with rotational flow cells.
3. Event Frequency - Number of events per hour.
4. Event Duration - Percent of time in specified event type.
5. Average Event Period - Average length of specified event type.
6. Loop Flow Change - Percent change in loop flow.
7. Core Exit T/C - Maximum observed T/C change for event type.
8. RVLIS Pressure - Change in RVLIS pressures, A and B channels.

In Table 3-3, similiar conditions were grouped for direct comparison, e.g. the flat lower support plate/neutron pad plants are in Table 3-3(A) and the dome support plate/thermal shield style in Table 3-3(B). In Table 3.4, significant reactor design features are listed for the flat core support plate style plants.

A review of the data in Tables 3-2, 3-3, and 3-4, provides several significant observations which when combined with subscale hydraulic test data, produces a better understanding of the plant events. The following are particularly pertinent:

1. Plants with identical reactor geometry; inlet nozzles, vessel downcomer, lower plenum, core support and lower core plate; are observed to have one of three conditions, no events, unsteady event, or nearly steady events.

2. The most frequent or dominant event, in each of the four units showing clear indications, Table 3-3(A), is the N-44/42 condition.

3. Of the four plants which show discrete events, the durations of the N-44/42 event varies from [ ]^{b,g} at Catawba 1 to [ ]^{b,g} at Callaway.

4. Of the flat support plate style plants, the anomaly has not been observed in units with 3.75 inch diameter BMI columns and 3.0 inch thick tie plates.

5. In the dome support/thermal shield plants, the dominant event is an N-43.

6. In the dome style plants, the events are not discrete but appear to be continually varying.

With the plant data indicated above and a series of subscale hydraulic test data sets, References 2 and 6, an empirical model of the flow anomaly was developed in order to relate the observed phenomena to plant configurations and operating conditions.

3.2.4 Empirical Model of Flow Anomaly

The development of rotational cells is believed to result from interacting flow phenomena initiated by several independent factors, a number of which are constant and at least one of which is variable from plant to plant. The constant factors tend to induce or enhance a condition in which the flow disturbance can develop. The variable factor serves to either suppress the rotational flow condition, cause it to be present but stable and non-observable by normal methods, or cause it to be intermittent, oscillatory, and observable as in Callaway, Wolf Creek, and the Catawba Units.

In terms of the reactor flow path, the initial factor appears to be the pairing of the reactor vessel inlet nozzles in the 4-loop configuration which results in a circumferentially non-uniform velocity in the vessel/core barrel downcomer. An example of this non-uniform velocity distribution is shown in subscale hydraulic test data, Figure 3-1. The peak downcomer velocities occur between the inlet nozzles on the vessel 90 degree and 270 degree axes. In the newer style of 4-loop plants with neutron pads, the high velocity flow in this region has a direct path to the lower plenum unimpeded by neutron pads or radial keys, as shown in Figure 3-2. This non-uniformity produces a preferential flow penetration into the limited volume at the bottom of the lower plenum which because of the hemispherical geometry involved leads to the development of rotational flow cells. The high velocity flow reaches the bottom of the plenum while the lower velocity flow turns more rapidly immediately beneath the core support plate. The result is the development of a rotational flow condition.

Lower plenum rotational flow cells have been observed in subscale hydraulic tests, as indicated in data from References 3 and 4 and reported by observers of sub-scale testing at the Centre de Cadarache, France and at the Takasago National Institute, Japan. In tests with open plenums, high rotational velocities develop, causing low pressure regions in the plenum and locally reduced flow through the lower core plates. Two examples of this effect on core inlet flows as may be observed in Figures 3-4 and 3-5.

A second factor, which can accentuate or reduce the inlet nozzle and downcomer effects, is the geometry of the internals structures in the lower plenum. The shape and placement of the lower tie plate which connects the BMI (Bottom Mounted Instrumentation) columns is believed to be particularly significant. Depending upon shape and position, it can influence formation of the rotational flow cells, as has been shown in Reference 4. A correlation of data on this effect from Reference 4 is provided in Figure 3-6. As shown, if positioned close the vessel lower head, the tie plate will disrupt formation of rotational cells. Conversely, at a sufficient distance from the lower head, it will not significantly influence their formation. In the Westinghouse 4-loop design, the lower tie plate is non-symmetric, Figure 3-3, and appears to favor vortex formation in certain vessel quadrants. In particular, the vessel 0 to 90 degree quadrant (N-44) and 270 to 0 degree quadrant (N-42) have tie plate geometries which provide more flow area into the bottom of the plenum. Other things being equal, such as loop flows, the tie plate geometry favors the formation of two vortices in the above indicated quadrants. This has been observed in the Catawba Unit 1 which has a nearly stable vortex condition [ ]^{b,g} in quadrants N-44 and N-42, at Wolf Creek which has the same condition for [ ]^{b,g} of the time, and at Callaway for [ ]^{b,g} of the time. As noted in Table 3-3(A), the excore detectors labeling is reversed at Catawba 2 and the 0 to 90 degree quadrant corresponds to the N-41 detector and the 270 to 0 degree quadrant corresponds to the N-43 detector. Thus, the dominant event type at Catawba 2, and N-41/43, is in the identical lower plenum quadrants as the other three plants.

The third factor considered, is a variable structural density in the lower plenum hardware. As indicated in Table 3-4, there are two thicknesses of tie plates and, similiarly, two BMI column diameters in Westinghouse 4-loop plants. In general, the thinner tie plates are matched with the smaller column diameters. Sub-scale hydraulic test data has shown that reduced structural density in the lower plenum increases the probability of vortex formation. Hydraulic testing of the Advanced Pressurized Water Reactor (APWR) at Takasago, Japan showed strong rotational cells which significantly affected the core inlet flows when no hardware was installed in the plenum, Figure 3-4. Conversely, the installation of BMI columns and tie plates suppressed the effect of the vortices. While still visable with dye and air injection, no significant effect on the core inlet flow was observed. This effect is observed in the plant data, where none of the plants with the higher structural density have reported the anomaly.

The fourth factor, found to be involved, is the loop to loop flow imbalance which is variable and unique to each plant. All plants are likely to have loop flow imbalances of [$]^{b,g}$ of the average loop flow due to differences in pump heads and loop component flow resistances. The newer 4-loop plants are also considered to have an inherent difference between adjacent loop flows of as much as [$]^{b,g}$ due to the difference in direction of the rotating flow at the suction of adjacent pumps. The plant to plant variations in the magnitude and the relationship of the flows entering the lower plenum are believed be responsible for the plant to plant differences in vortex behavior. In the case of equal loop flows, each loop flow tends to be contained in its respective quadrant of the lower plenum, providing a set of flow boundaries conducive to vortex formation. When the imbalance becomes significant, the high flow (high momentum) loop(s) tends to displace the flow from other loops in the bottom of the lower plenum. The respective flow boundaries in the lower plenum are no longer separated by the planes thru the 0-180 and 90-270 degree axes. The resulting flow regions are no longer quarter vessel symmetric and appear less likely to produce the vortex conditions.

One flow anomaly correlation, based on calorimetric loop flow measurements, is shown on Figure 3-7 where the fractional period of the N-44/42 vortex is related to the loop flow imbalance of opposite loops. The four flat core support plate plants, which have clear indications of the anomaly, show a trend of decreasing vortex activity with increasing opposite loop flow imbalances. When the imbalance is small, i.e. <1%, stable vortices exist and the tie plate effects dominate, as in Catawba 1. When the imbalance is greater, as at Wolf Creek [ ]^{b,g} and Callaway [ ]^{b,g}, the stability of the vortices is disturbed and the duration of the vortices is diminished. When the imbalance is even larger [ ]^{b,g}, the flow conditions at the bottom of the plenum appear to suppress formation of vortices, as at Byron 2.

The correlation suggests that vortex activity is eliminated with a [ ]^{b,g} flow imbalance, however, the uncertainty in the measured flows is significant and the [ ]^{b,g} threshold has corresponding uncertaintity. The plants shown with solid symbols in Figure 3.7 provide additional data that lower internals geometry has a major influence on the occurence of the vortices. All plants in this group have 3.75 inch diameter BMI columns and all, except Byron 1, have 3.0 inch thick tie plates. While the configuration of this hardware is identical to the previous group, none of the plants with the larger columns are observed to have the vortex conditions. This is consistent with sub-scale hydraulic test data which shows reduced vortex effects with increasing lower plenum structural density.

The above discussion has been related to 4-loop plants with the flat lower core support plate type of lower internals. However, as is indicated in Table 3-3(B), some anomaly type events occur in one dome core support/plate style plant which has no tie plates. The paired inlet nozzles, radial key positions, and hemispherical plenum geometry are apparently sufficient to induce a rotational flow condition with this type of support plate. With the absence of the tie plates, the N-44 and N-42 are no longer the dominant quadrants and the rotational cell is located primarily in the N-43 quadrant. As may be observed in the raw data (see Appendix), the characteristics of this plant type are distinctly different and lack the abrupt formation and breakup of the rotational cells seen in the flat support style plant. In addition, the magnitude of the parameter variations is substantially less than observed for the flat support style plants, with no changes detectable in RCS loop flow rates.

3.2.5 Summary and Conclusions

In summary, it appears that the Hydraulic Flow Anomaly can be attributed to multiple rotational flow cells (vortices) which develop on the lower plenum of 4-loop plants. In most cases, the events can be attributed to a pair of vortices. The geometry and flow conditions of 2-loop and 3-loop plants appear to preclude similiar phenomena. In 4-loop plants with flat core support plates, the development of these cells is attributed to a combination of several factors: 1) the pairing of the inlet nozzles producing a local peak in downcomer flow velocity; 2) the position of the neutron pads and radial keys which provide a preferred flow path to the lower plenum; 3) a non-symmetric tie plate design in the lower plenum which makes the N-44 and N-42 quadrants preferred locations for the rotational flows; 4) a reduced lower plenum structural density (thinner tie plates and BMI columns) which reduces flow energy dissipation rates, and 5) plant specific loop to loop flow imbalances which modulate the tie plate and column effects and produce plant to plant variations in the observed phenomena. In one 4-loop plant with the dome core support plate, the paired inlet nozzles, radial key positions, and loop to loop flow imbalances appear sufficient to produce a similar phenomena but with a substantially smaller magnitude than in the flat core support plants, a more continuously varying characteristic, a single quadrant vortex, and no impact on RCS loop flow.

TABLE 3-2

SUMMARY OF PLANT DATA

CATAWBA 1

Data Available: Set 1

- Two sets of 40 minutes each; 16:10 to 16:50 on 1/5/87 and 10:51 to 11:31 on 1/6/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- Loop average temperatures.
- Core exit temperatures; K05 and D11.
- Loop delta temperatures; in % of full power.

b,g

[ ]

Data Available: Set 2

- Tape recorded data, collected 3/31/87 thru 4/1/87.

Data Available: Set 3

- Strip chart data, collected 5/14/87 thru 5/18/87.

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

CATAWBA 2

Data Available: Set 1

- One set of 40 minutes; 09:29 to 10:09 on 1/6/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- Cold leg temperatures.
- Core exit temperatures; L14 and C06.

b,g

Data Available: Set 2

- Tape recorded data, collected 5/13/87 thru 5/14/87.
- Strip chart data, collected 5/13/87 thru 5/14/87.

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

WOLF CREEK

Data Available:

- Two sets of 60 minutes each; 03:31 to 04:33 on 1/1/87 and 22:10 to 23:15 on 1/1/87. (Also several sets of N-1 and N-2 loop data.)
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- RVLIS A and B pressures.
- Core exit temperatures; L12, E12, and E4.

b,g

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

b,g

[ ]

CALLAWAY

Data Available:

- Approximately 34 hours of data; 20:00 on 12/10/86 to 06:00 on 12/12/86.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- RVLIS A and B pressures.
- Core exit temperatures; C4, E4, G4, J4, L4, N4, C6, G6, J6, N6, E8, C10, G10, N10, C12, E12, J12, L12, and N12.

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

b,g

DIABLO CANYON 2

Data Available:

- One set of 4 hours data; 13:34 to 17:34 on 4/2/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- RVLIS 201 and 204 pressures.
- Core exit temperatures; E2, L2, H5, P5, E6, H8, E10, J10, C12, and K13.

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

Review of the four hours of data provides no indications in the excore data, the loop flow data, or the core exit thermocouple data of the transitional fluctuations. The excore data visually indicates equal noise on the four channels, similar to the no vortex noise pattern observed at Callaway and Wolf Creek during the non-event periods. [

]^{b,g}

MCGUIRE 1

Data Available: Set 1

- Excore and incore data, recorded 10/9/86.

Data Available: Set 2

- Four sets of 40 minutes each; 10:09 to 10:49, 10:50 to 11:30, 11:30 to 12:10, and 12:10 to 12:50 on 1/14/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- Loop average temperatures.
- RVLIS A pressures, in % full scale.
- Loop delta temperatures; in % of full power.

Review of the three hours and 40 minutes of data provides no indications in the excore data, the loop flow data, or the core exit thermocouple data of the event type fluctuations. The excore data shows generally equal noise on the four channels, which is similiar to the no vortex noise pattern observed at Callaway and Wolf Creek during the non-event periods. Thus, McGuire 1 is considered to have no vortex conditions in the lower plenum.

TABLE 3-2 (continued)

SUMMARY OF PLANT DATA

MCGUIRE 2

Data Available: Set 1

- Excore and incore data, recorded 10/10/86.

Data Available: Set 2

- Four sets of 40 minutes each; 09:56 to 10:36, 10:38 to 11:18, 11:18 to 11:58, and 11:55 to 12:35 on 1/14/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- Loop average temperatures.
- RVLIS A pressures, in % full scale.
- Loop delta temperatures; in % of full power.

The data from McGuire 2 is identical to that of McGuire 1. A review of the three hours and 40 minutes of data provides no indications in the excore data, the loop flow data, or the core exit thermocouple data of the event type fluctuations. The excore data shows generally equal noise on the four channels, which is indicative of no vortex. Thus, McGuire 2, like McGuire 1, is considered to have no vortex conditions in the lower plenum.

MILLSTONE 3

Data Available:

- One set of 4 hours; 11:00 to 15:00 on 1/8/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- Core exit temperatures; L12, E12, L4, E4, N10, C10, N6, and C6.

A review of the four hours of data provides no indications in the excore data, the loop flow data, or the core exit thermocouple data of the transitional fluctuations. The excore data visually shows minimal variation in excore noise levels. Thus, Millstone 3 is considered tc have no vortex conditions in the lower plenum.

BYRON 1

Data Available: Set 1

- One hour of data; Received on 1/7/87
- Excore power; N-41, N-42, N-43, and N-44.

Data Available: Set 2

- Three sets of 40 minutes of data; Collected on 6/24/87
- Excore power; N-41, N-42, N-43, and N-44
- Incore data; L5, E5, L11, and E11

A review of the data provides no indications of the transitional fluctuations. Spectral analysis of the incore and excore data shows similar noise levels in the four quadrants for the low frequency bands. Thus, Byron 1 is considered to have no vortex conditions in the lower plenum.

BYRON 2

Data Available:

In excess of two hours of data; Taken on 6/5/87
Excore power; N-41, N-42, N-43, and N-44.
Incore detectors; E11, L11, E5, L5, K12, J10, L10, N14, H11, C8, D8
H3, H13, G9, B8, and N8.
Loop flow rate from elbow taps.

Review of two hours of data provides no indications in the excore data, the loop flow data, or the incore flux data of the transitional conditions seen at Callaway. The excore data shows equal noise on the excore four channels,

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

which is characteristic of the no vortex noise pattern observed at Callaway and Wolf Creek during the non-event periods. Thus, Byron 2 is considered to have no vortex conditions in the lower plenum.

<u>SOUTH TEXAS</u>

Data Available:
- One hour of data; 2/22/87, Hot Functional Tests.
- Loop flow rate from elbow taps.

b,g

<u>TROJAN</u>

Data Available: Set 1

Data recorded 1/30/80 and 10/15/80.

Excore power; N-41, N-42, N-43, and N-44.

Data Available: Set 2

One hour of data recorded on data logger, April, 1987

Excore power: N-41, N-42, N-43, and N-44.

Loop flow rate form elbow taps.

RVLIS pressure.

Core exit thermocouples; K11, E05, and E10.

Review of the data shows no indications in the excore flux of the transitional conditions seen at Callaway, neither in the data collected in 1980 nor in the data collected in 1987. Note that this data represents conditions before and after the conversion of Trojan from a downflow to an upflow baffle/barrrel system. The data also shows equal noise on the excore channels, which is characteristic of the no vortex noise pattern observed at Callaway and Wolf Creek during the non-event periods. Thus, Trojan is considered to have no vortex conditions in the lower plenum.

INDIAN POINT 2

Data Available:

A - Four NIs Power Range Excore Tapes, collected 4/20/83, 11/30/84, 7/22/86 and 8/14/86.

B - One set of data with a length of 4 hours and 20 minutes collected 1/19/87. Data printout frequency was 5 seconds.

NIS Power Range Excore Detectors
Incore Thermocouples E2, E14, L2, and L14
RCS Loop Flows
RVLIS - Chs A & B
Steam Generator Pressures
RCS Loop T-AVG
RCS Loop Delta-T
Generator Load
Controlling Rod Bank Position

C - 44 sets of data with varying lengths between 8 to 60 minutes (a total of 9 hours) collected over a period of 2/28/87 to 3/10/87. All parameters were collected at a 2 second scan rate. Data printout frequency varied from 2 seconds to one minute averaged values.

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

INDIAN POINT 2 (cont.)

NIS Power Range Excore Detectors
Incore Thermocouples M5, E11, F5, L11, E2, and B6
RCS Loop Flows
RVLIS - Ch A
Steam Generator Pressure
RCS Wide Range Loop T-Cold and T-Hot

b,g

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

INDIAN POINT 2 (cont.)

b,g

[ ]

INDIAN POINT 3

Data Available:

- One set of data of 2 hours in length; collected on 4/23/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- Core exit temperatures; B10, F12, J10, and K03.

The review of the IPP-3 data shows that no flow disturbance is present. The step changes in data level do not appear, neither as at Callaway and Wolf Creek nor more gradually as observed at IPP 2. The excore flux data indicates essentially equal noise on all four channels. Thus, IPP 3 is considered to have no vortex conditions in the lower plenum.

TABLE 3-2 (continued)
SUMMARY OF PLANT DATA

DIABLO CANYON 1

Data Available:

- One set of 3.25 hours of data; 9:52 to 14:07 on 3/4/87.
- Excore power; N-41, N-42, N-43, and N-44.
- Loop flow rate from elbow taps.
- RVLIS 201 and 204 pressures.
- Core exit temperatures; E2, L2, D4, D7, G8, D9, J11, F12, L12, and H13.

Review of the four hours of data provides no indications in the excore data, the loop flow data, or the core exit thermocouple data of the transitional fluctuations. The excore data visually indicates equal noise on the four channels, similar to the no vortex noise pattern observed at Callaway and Wolf Creek during the non-event periods. However, plant computer processed data has not been shown to accurately depict the neutron noise in the appropriate frequency range and direct neutron noise measurement and analysis would help to verify the no vortex condition.

TABLE 3-3(A); MATRIX OF PLANT CONDITIONS
FLAT LOWER CORE SUPPORT PLATE
LOWER INTERNALS DESIGN

Plant	Operating* Condition	Event Type	Event Frequency Events/hr	Event Duration % Time	Avg. Event Period Min.
Catawba 1	Steady Event*	N-44/42 N-43/42/41 N-43/42	[		]b,g
Catawba 2	Unsteady Event	N-41/43 (N-44/42)**			
Wolfcreek	Unsteady Event	N-44/42 N-44/43/42 N-43/41			
Callaway	Unsteady Event	N-44/42 N-43/41 N-43/42			
Diablo Canyon Unit 2	No Event				
McGuire 1	No Event				
McGuire 2	No Event				
Millstone 3	No Event				
Byron 1	No Event				
Byron 2	No Event				
South Texas	Possible Unsteady Event	***	***	[ ]b,g	***
Trojan	No Event				

* Condition considered steady if duration is greater than 95% of time

** Equivalent to N-44/42 in other plants.

*** Insufficient or no data.

TABLE 3-3(B); MATRIX OF PLANT CONDITIONS
DOME LOWER CORE SUPPORT PLATE
LOWER INTERNALS DESIGN

Plant	Operating* Condition	Event Type	Event Frequency Events/hr	Event Duration % Time	Avg. Event Period Min.
Indian Point Unit 2	Unsteady Event	N-43 N-42 N-44	[		] b,g
Indian Point Unit 3	No Event				
Diablo Canyon Unit 1	No Event				

* Condition considered steady if duration is greater than 95% of time
** Indeterminate from available data.

TABLE 3-4; MATRIX OF REACTOR DESIGN FEATURES
FLAT LOWER CORE SUPPORT PLATE
LOWER INTERNALS DESIGN

Plant	Operating* Condition	BMI Col. Diameter (Inches)	Tie Plate Thickness (Inches)	Baffle/ Barrel Design	RCP Flow Splitter
Catawba 1	Steady Event*	3.00	2.00	Up Flow	No
Catawba 2	Unsteady Event	3.00	2.00	Up Flow	No
Wolfcreek	Unsteady Event	3.00	2.00	Up Flow	No
Callaway	Unsteady Event	3.00	2.00	UpFlow	No
Diablo Canyon Unit 2	No Event	3.75	3.00	Down Flow	Yes
McGuire 1	No Event	3.75	3.00	Down Flow	No
McGuire 2	No Event	3.75	3.00	Down Flow	No
Millstone 3	No Event	3.75	3.00	Up Flow	No
Byron 1	No Event	3.75	2.00	Up Flow	No
Byron 2	No Event	3.00	2.00	Up Flow	No
South Texas**	Possible Unsteady Event	3.00	3.00	Up Flow	No
Trojan	No Event	3.75	3.00	Conv. Up Flow	No

* Condition considered steady if duration is greater than 95% of time

** South Texas lower core support/core plate design varies significantly from other units.

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-1 APWR Downcomer Velocity Distribution

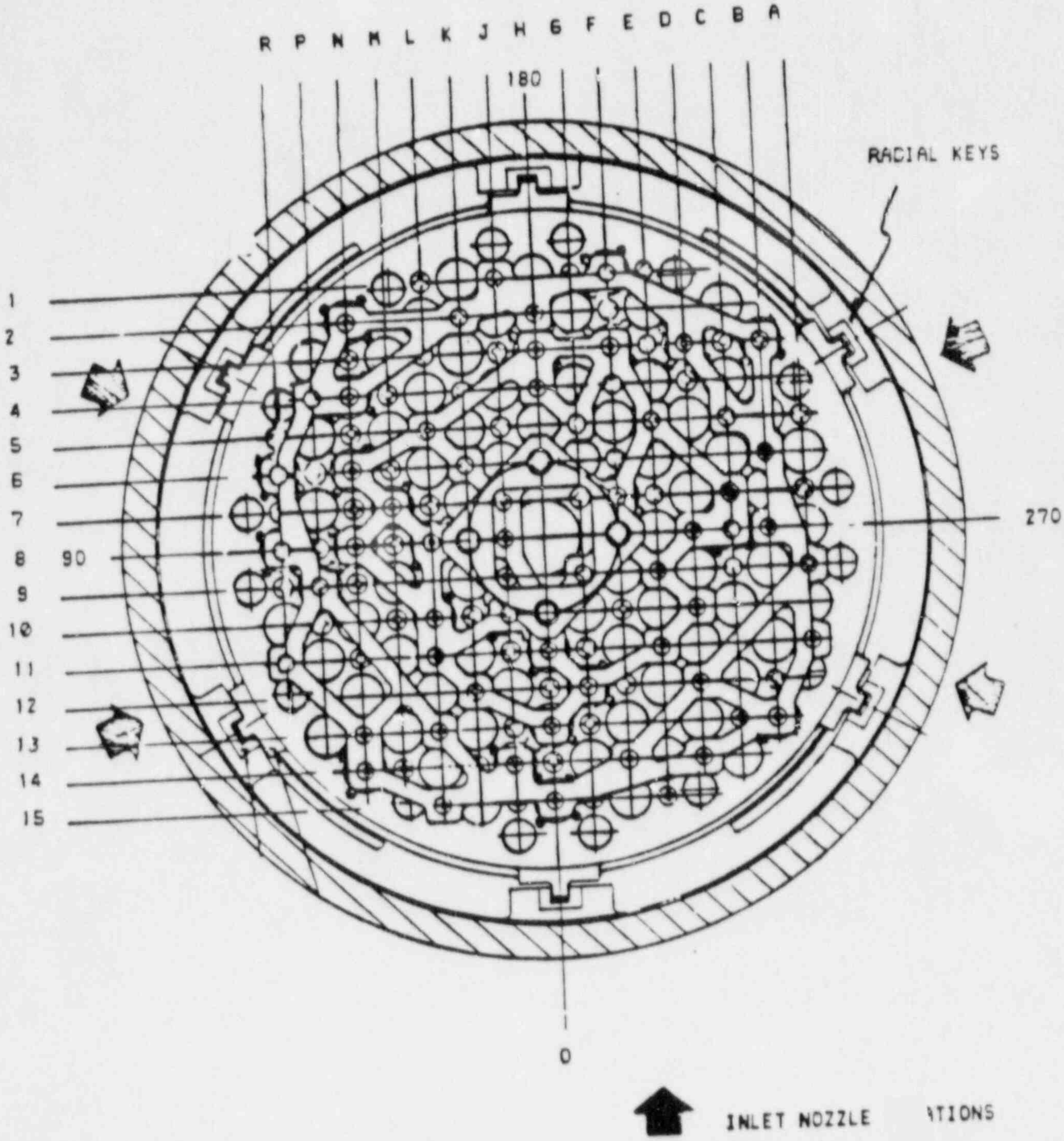

Figure 3-2 Downcomer and Lower Plenum Inlet Flow Areas

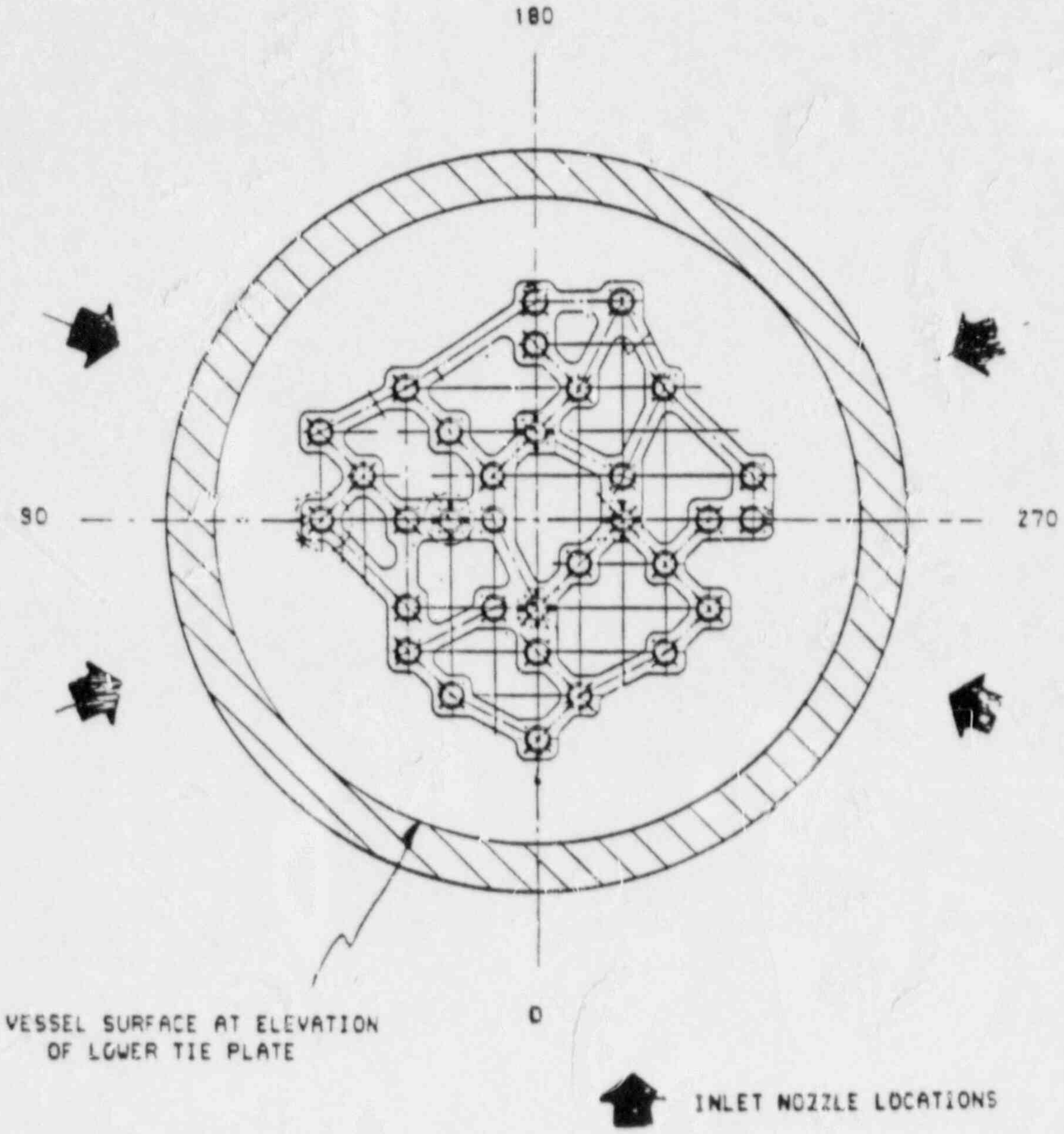

Figure 3-3 Reactor Vessel Head/Lower Tie Plate Flow Area

b,g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-4 APWR Core Inlet Flow Distribution Without Lower Plamum Hardware

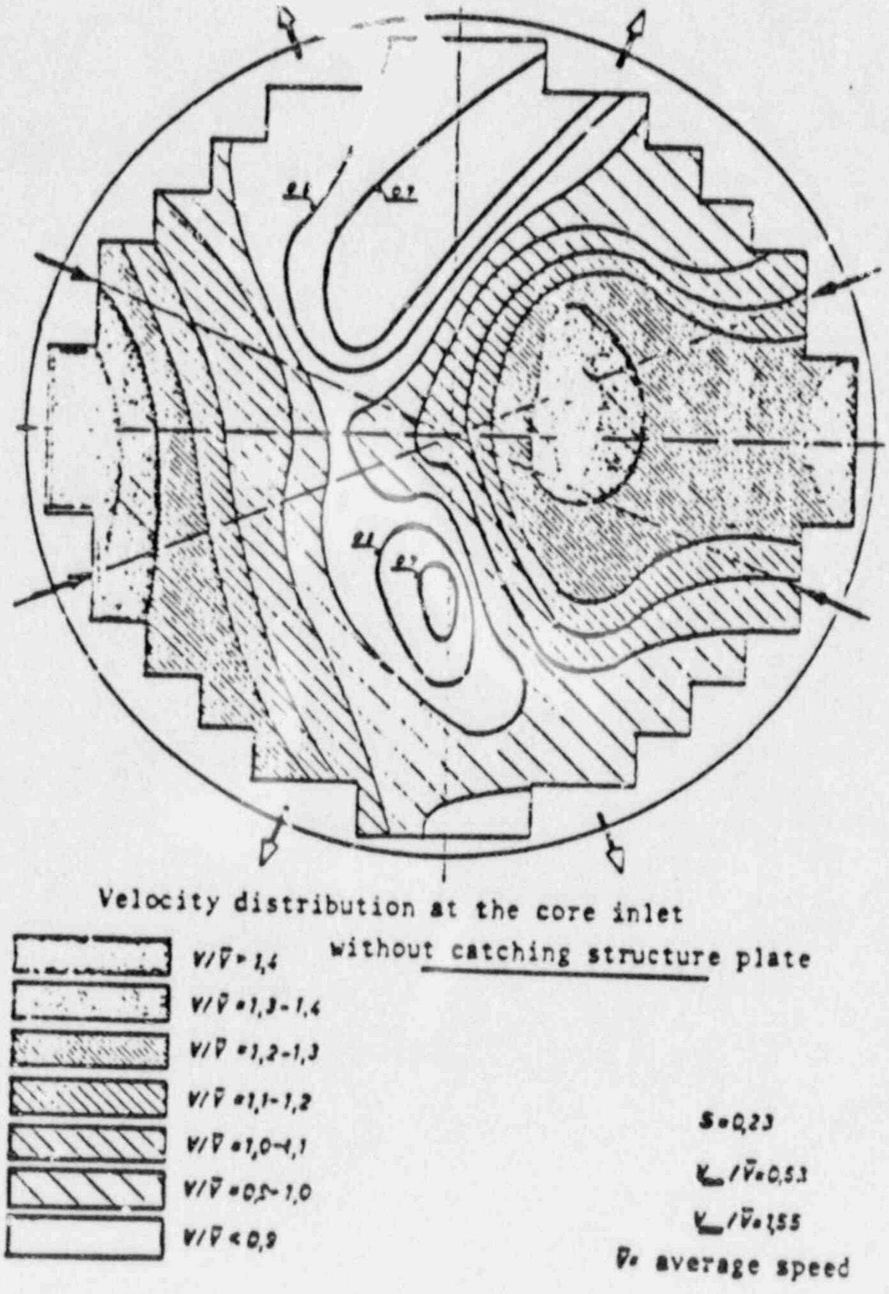

Figure 3-5 Siemens Core Inlet Flow Distribution
Without Lower Plemum Hardware

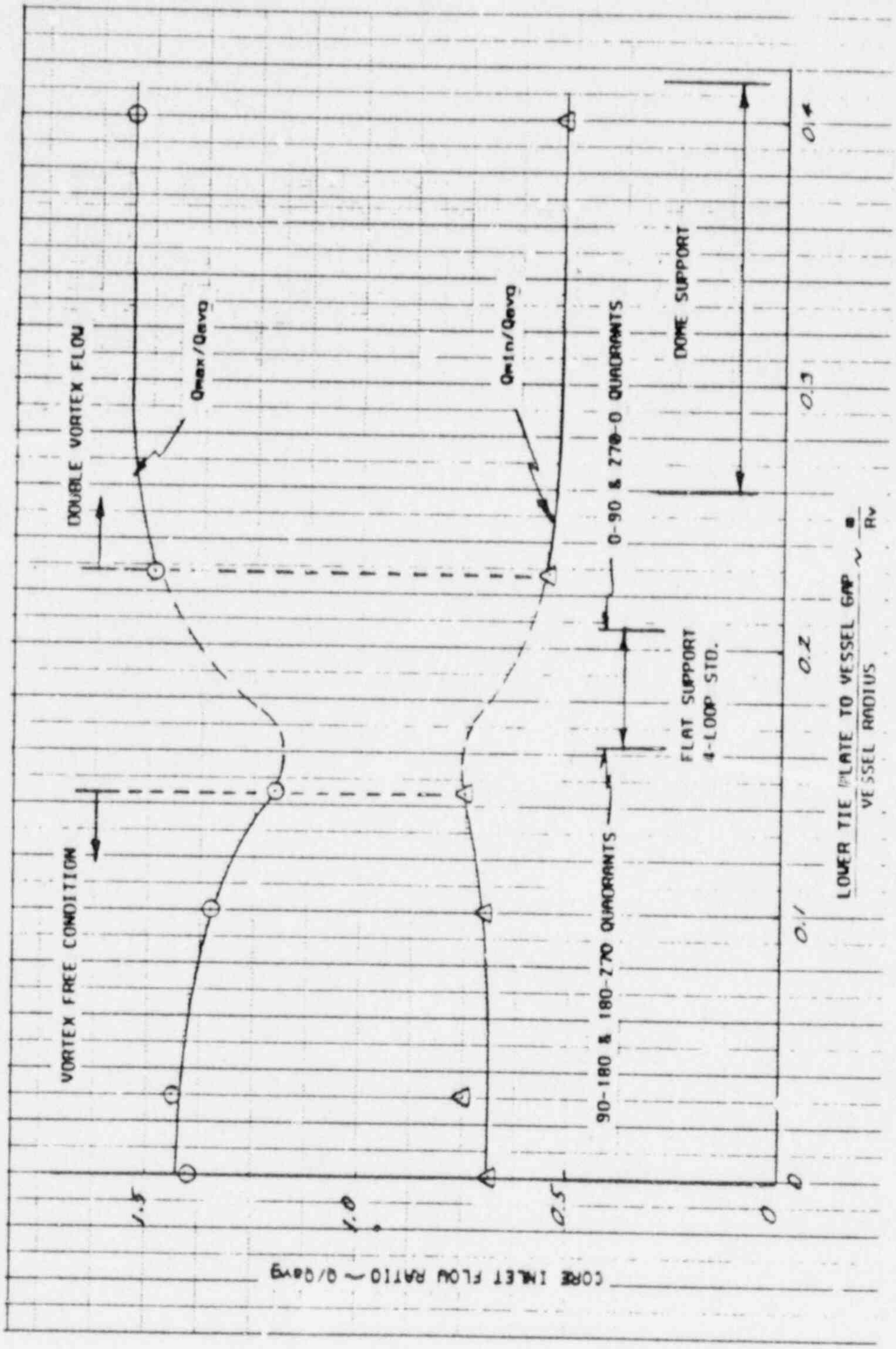

Figure 3-6 Core Inlet Flow Ratios

Figure 3-6 Core Inlet Flow Ratios

b, g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-7 Fraction of Time with Steady Vortex

3.3 LARGEST OBSERVED PERTURBATIONS

3.3.1 Nuclear Signals

Table 3-5 gives the largest signal perturbations that have been observed at the plants where flow anomaly transitions have been seen. The largest ex-core change is [ ]b,g although none of the measurements are for the end of core design life where the more negative moderator temperature coefficient could cause this magnitude to increase. The largest in-core deviation is [ ]b,g although this may also increase with a more negative moderator temperature coefficient near the end of core design life.

3.3.2 Loop Flows

The loop flow changes range from [ ]b,g except for multiple event conditions (which have been observed very infrequently) where loop flow decreases of [ ]b,g have been observed.

3.3.3 Core Exit Thermocouple Temperatures

The thermocouple temperature changes measured at most plants are [ ]b,g or smaller, except at Catawba Unit 1, where thermocouple M-11 showed a [ ]b,g increase and thermocouple K5 showed an increased of [ ]b,g

Thermocouple M-11 was erratic and, therefore, suspect in tests performed on 3/31/87 and 4/01/87. Data from 5/14/87 on Figures 3-8 and 3-9 show a lower response to the N44 anomaly and a high noise level for this thermocouple. Thus the [ ]b,g change observed for thermocouple M-11 at Catawba 1 is suspect.

Thermocouple K5 shows a [ ]b,g increase during the N43/41 anomaly as seen on Figures 3-8 and 3-9. The response of this thermocouple to N44 anomaly dropouts is small [ ]b,g as seen on Figure 3-8. This thermocouple is in the core quadrant adjacent to ex-core detector N41 at Catawba Unit 1.

3.3.4 RVLIS Differential Pressure

The largest changes in RVLIS differential pressure is [ ]b,g psi drop in the B RVLIS channel at Wolf Creek and [ ]b,g psi drop at Callaway. In these two plants the B RVLIS channel lower pressure tap is at core location L-11, which is at or near the suspected vortex center in the N44 core quadrant.

The drop observed at Indian Point Unit 2 is substantially lower at [ ]b,g psi. The RVLIS lower pressure tap is at core location E-5, which is near the suspected vortex center in the N43 core quadrant.

RVLIS data is not available from the two Catawba units.

TABLE 3-5; LARGEST OBSERVED PERTURBATIONS

Plant	Anomaly Type	Ex-Core Deviation % (Detector)	In-Core Deviation At Core Mid-Height % (Location)	Loop Flow Change %	Core Exit T/C Change °F	RVLIS B/A Pressure Change psi
Flat Core Support Type Plants						b,g
Callaway	N-44/42					
	N-43/41					
	N-43/42					
Wolf Creek	N-44/42					
	N-44/43/42					
	N-43/41					
Catawba 1	N-44/42					
	N-43/42/41					
	N-43/42					
Catawba 2	N-41/43					
	N-41/42					
Dome Core Support Type Plants						
Indian Point 2	N-43	0.5% (N-43)	No Data	0.0%	0.6 F	-1.2 psi

Notes: (1) Mixed anomaly transition from N44 to N43/41
(2) Suspect thermocouple M-11

b.g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-8 Catawba 1 NIS and Thermocouple Data

b,g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-9 Catawba 1 NIS and Thermocouple Data

3.4 THERMAL/HYDRAULIC AND NUCLEAR EVALUATION

For the thermal-hydraulic evaluation of the Callaway flow anomaly which was presented to the NRC on January 16, 1987 (Reference 1), [

]^{a,c} The updated analytical model reported here includes several refinements which have been made to enhance the agreement between the calculated results and the test data. The updated model is discussed below and a summary of model assumptions is given in Table 3-6(A).

3.4.1 [ ]^{a,c} Simulation

The initial evaluation of the flow anomaly which was presented assumed a [ ]^{a} Although this assumption results in a [ ]^{c} the results [ ]^{c} and the model was judged to be conservative with respect to effect on DNB and cross flow in the fuel rod bundles. The current evaluation uses [ ]^{a,b,c} flow model based on [ ]^{a,b,c} model test data (refer to Section 3.2 for more information about the [ ]^{a,b,c} tests). The [

]^{a,b,c} as illustrated by the normalized inlet flow deficiency map shown in Figure 3-10. The flow reduction is [

]^{a,b}

3.4.2 Power Distribution

The initial analysis of the flow anomaly (Reference 1) used a [ ]^{a} power distribution to determine the degree of flow maldistribution at the core inlet, whereas the analysis presented herein used the [ ]^{a} power distribution for Callaway Cycle 2 so that a more realistic correlation of analysis results with measured data is obtained. This power distribution exhibits a [

]^{a}

A more realistic [ ][a] was used replacing the [ ][a] (Reference 10). Both [ ][a] are shown in Figure 3-11. In addition, the power reduction due to [ ][a] rather than uniformly throughout the perturbed region.

3.4.3 Simulation of N-44/42 Event

The measured core exit thermocouple temperature changes during a N-44/42 event, which are shown in Figure 3-12, were reviewed to determine the location of a [ ][a,c] It was concluded that the 'worst' case core inlet flow distribution would be simulated by modeling the [ ][c]

A [ ][a] model was developed to [ ][a] as described in Section 3.4.1, [ ][c] Alternate models, simulating [ ][c] were developed and used to confirm that the [ ][c] is conservative. For each model, the [ ][a]

3.4.4 THINC-IV Results

A series of [ ][a] runs were made with the [ ][a] model described above using best estimate operating conditions for Callaway. [ ][a,c]

[

]^{a}c The results are summarized in Table 3-6 (B) which includes a comparison of the calculated temperature increases at selected locations with the observed data. A core map of the calculated exit temperature changes is shown in Figure 3-13.

The [ ]^{a,c} results show that [

]^{a,c}

The [ ]^{c} in the perturbed region is shown in Figure 3-15 which plots the [

]^{c} for several of the affected channels. [

]^{c}

The lower plenum flow disturbance results in [

]^{a,c}

3.4.5 Coolant Temperature Changes vs Neutron Flux Measurements

Two of the primary indications of the flow anomaly are an increase in core coolant temperatures in the perturbed region as detected by the core exit thermocouples and a corresponding reduction in neutron fluxes as measured by in-core and ex-core NIS instrumentation. The observed decrease in neutron flux is due to the reactivity feedback response to the coolant temperature increase (moderator density decrease). The [ ]^{a} were used to estimate the corresponding reductions in neutron flux. The axial variation of temperature change at representative locations is shown in Figure 3-16. The axial average coolant temperature increase for each assembly in the perturbed region was input to [ ]^{a} calculations to determine the [ ]^{a} for each assembly. The axial distribution of flux change was obtained from the [

]^{a} The neutron flux changes at core locations [ ]^{a,c} predicted from the [ ]^{a,c} changes are compared to the measured values in Figure 3-17. Considering the uncertainties involved in the neutronics model and assumptions as well as the measurement uncertainties, this comparison clearly shows a correlation between the observed neutron flux change and the calculated coolant temperature increases.

3.4.6 Conclusions

A [ ]^{a,b} model based on [ ]^{a,b} flow data was used to evaluate the RCS Flow Anomaly. This model is judged to be an advancement to the [ ]^{a,c} models used earlier. It was also found that the [ ]^{a,c} is an important factor to be considered in the search for a model to simulate the actual disturbance. On the basis of this study, it was concluded that the flow anomaly N-44/42 event is simulated best as [ ]^{a,b,c} with inlet flow maldistribution based on [ ]^{a,b,c} test data. This model is recommended for use in safety evaluations for plants which have the RCS Flow Anomaly.

TABLE 3-6 (A)

ASSUMPTIONS USED IN THINC-IV FLOW ANOMALY CALCULATION

Initial Evaluation (January, 1987)	Updated Evaluation
- [ ]^{a,c} assemblies	- [ ]^{a,c} assemblies
- [ ]^{a,c} flow reduction	- [ ]^{a,c} flow reduction based on APWR scale model test data
- Power reduction [ ]^{a,b,c}	- Power reduction [ ]^{a,b,c}
- [ ]^{a} power distribution [ ]^{a}	- [ ]$_{a}$ power distribution for Callaway Cycle 2
- [ ]^{a} axial power shape (Reference 10)	- [ ]^{a} power shape for Callaway Cycle 2
- Reactor flow [ ]^{a,b}	- Reactor flow [ ]^{a,b}
- [ ]^{a,b} ΔT increase	- [ ]^{a,b} ΔT increase

RESULTING FLOW MODELS:

Initial Evaluation (January, 1987)	Updated Evaluation
- [ ]^{a,c} flow reduction or - [ ]^{a,c} flow reduction	- [ ]^{a,c}

TABLE 3-6 (B)
SUMMARY OF VORTEX FLOW MODEL RESULTS

Parameter	Calculated	Measured	
[ ]a,c	[ ]a,c		
Core inlet flow reduction:			
Average [ ]a,c	[ ]a,c		
Peak [ ]a,c	[ ]a,c		
Temp. change (Deg F):	D-Temp	D-Temp	D-Flux

[ ]a,b,c

b,g

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Figure 3-10 Normalized Core Inlet Flow Deficiency in Vortex Region

b,g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-11 Callaway Cycle 2 Average Axial Power Shape b,9

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-12 Measured Exit T/C Temperature Change from Callaway Data

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-13 Calculated Core Exit Temperature Changes

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-14 Effect of Crossflow on Delta-T Between Assemblies

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-15 Flow Recovery in Perturbed Region

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-16 Axial Variation of Temperature Change b,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-17 Comparison of Measured to Predicted Change in Neutron Flux (Callaway Data)

3.5 HYDRAULIC LOADS

3.5.1 Introduction

In order to evaluate the loads induced by the RCS Flow Anomaly on 4-loop reactor lower internals, a method has been developed to determine rotational cell (vortex) velocities within the lower plenum. The observed RVLIS pressure changes in the lower plenum were used to determine the vortex strength and incorporated into rotational flow equations to provide vortex velocity estimates. RVLIS data from Callaway and Wolf Creek was available for evaluation of flat support plate plant velocities and similar data was available from Indian Point 2 for evaluation of dome support plate plant velocities.

3.5.2 Methodology

Based upon the plant data collected to date and the available subscale model data, an empirical model of the flow conditions in the lower plenum has been developed, Section 3.2. In that model, two rotational cells are shown to be the usual condition during an event. These cells are located most frequently in the 0 to 90 and 270 to 0 degree quadrants, corresponding to the N-44 and N-42 excore detectors. As indicated in Table 3-3, the duration of this condition varies from 10% of the time at Callaway to 96% of the time at Catawba 1; essentially a constant condition. Based upon the [ ]^{b,g} Test flow visualization, the flow rotation is expected in the directions shown in Figure 3-18. For structural analysis purposes of BMI columns and tie plates, the flow pattern in Figure 3-19 can be assumed with the flow velocities in the 90 to 180 and 180 to 270 degree quadrants evaluated by standard methods.

The approach to obtaining conservative estimates of velocity within the rotational flow cells is based upon using analytical vortex flow expressions for static pressure and velocity and determining the related coefficients from "measured" data. The analytical expressions were developed for the potential flow and viscous flow regions of the vortex from a combination of vortex theory, Ref. 8 and physical arguments. The velocity field and pressure field equations were then compared to experimental data, Reference 8, on a vortex flow condition in a free jet. The comparison was found to be reasonable and supported use of the analytical expressions for estimating plant velocities. Two coefficients were required for the equations, 1) the outer radius of the viscous region and 2) the coefficient of rotation (or vortex strength). The radius of the viscous region was estimated from a combination of physical arguments and plant data. The strength of the vortex was estimated through the pressure equation and the observed RVLIS pressure changes. The maximum observed pressure differential between the RVLIS A and RVLIS B channels, from Callaway and Wolf Creek data, was used for the evaluation. In addition, the center of the vortex was considered to be displaced from the RVLIS B tap so that the maximum pressure differential used exceeded the recorded values by [ ]^{a,c} The maximum observed pressure changes are listed in Table 3-7 together with the calculated peak rotational velocities.

3.5.3 Results and Conclusions

A non-dimensional plot of static pressure and vortex velocity is provided in Figure 3-19. Based on the Callaway and Wolf Creek RVLIS pressure changes, the maximum rotational velocity is [ ]^{a,c} Correspondingly, the maximum velocity for dome style plants, based on Indian Point 2 data, is [ ]^{a,c}

For estimating duration and frequency of events, a listing of applicable data is provided in Tables 3-3a and 3-3b.

TABLE 3-7; RVLIS PRESSURE DATA AND ESTIMATED
MAXIMUM ROTATIONAL VELOCITY

Plant Type	RVLIS B psi	RVLIS A psi	Total psi	Maximum Velocity ft/sec
				a,b
Flat Support (412)				
Flat Support (4XL)*				
Dome Support				

* Based in 412 (Callaway) pressures.

** Estimated from flat support plate ratio of RVLIS A to RVLIS B.

a,c

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Figure 3-18 Rotational Cell Nominal Locations and Flow Directions

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-19 Rotational Velocity and Static Pressure Ratios

3.6 LOW FREQUENCY NUCLEAR NOISE ANALYSIS

3.6.1 Summary and Conclusions

Analysis of low frequency nuclear noise from several 4-loop units shows that nuclear noise below 1 Hz is strongly influenced by the presence of the flow anomaly. At Callaway and Catawba Units 1 and 2, the low frequency nuclear noise below 1 Hz varies significantly between vessel quadrants when the flow anomaly is present. The pattern of high noise and low noise quadrants also varies with event type as shown in Figure 3-20. Conversely, at Callaway, when the flow anomaly was not present, and at McGuire Units 1 and 2, Byron Units 1 and 2, and at Trojan, the low frequency nuclear noise on the four ex-core detector channels, N41, N42, N43 and N44, is nearly equal in magnitude, Figures 3-20 thru 3-24. While the exact mechanism of this phenomenia is spectulative, the presence of the flow anomaly produces a characteristic signature in the pattern of the nuclear noise measured by both the incore and excore detectors. As shown in Figure 3-20, at Callaway the N-44/42 event has a noise signature characterized by low noise in the N41 and N44 quadrants, high noise in the N42 quadrant and an intermediate level in the N43 quadrant. The ratio of max. to min. noise is approximately 1.8. A similiar pattern and noise level ratio is observed at other plants during N-44/42 events; at Catawba 1 in Figure 3-29 and Catawba 2 in Figure 3-33. This signature provides a direct indicator of the presence of the N-44/42 event without the need to observe the transitional indications, i.e. steps in the data traces.

Given the very high fraction of the time that the Catawba Units spend with the flow anomaly present [ ][b,g] it is probable that some units with nearly equal loop flows will have a steady flow anomaly 100 percent of the time. The usual transitions will not be seen and the DC imbalance in the ex-core detectors will be calibrated out. The imbalances in core exit thermocouple data or power density measured with the movable detectors will be too small to detect. In this case, the characteristic signature of the low frequency nuclear noise can be used to determine the presence of a constant flow anomaly.

3.6.2 Ex-Core Nuclear Noise from Callaway

Figure 3-20 contains the low frequency nuclear noise from the bottom section ex-core detectors at Callaway for four flow anomaly event conditions. The left set of data is for the "no flow anomaly" condition which shows that the low frequency nuclear noise in the frequency bands 0.15 to 0.5 Hz and 0.15 Hz to 1.0 Hz are reasonably balanced between the four ex-core detectors with a maximum to minimum ratio of 1.2. The second set of data is for the N43/N41 anomaly where the dominant drop in DC signal during the transition to the anomaly is from the N43 ex-core detector. The low frequency nuclear noise in both frequency bands becomes significantly unbalanced with high noise levels in the N41 and N44 detector signals and low noise levels in the N42, and particularly in the N43 detector signals. The maximum to minimum ratio is 1.6, which is significantly higher than for the no anomaly condition.

The third set of data is for the N44/N42 anomaly where the main DC signal shift during the transition to the anomaly is from the ex-core detector N44. Again the low frequency nuclear noise shows a very strong unbalance with N42 and N43 having high noise levels and N41 and N44 having low noise levels. The maximum to minimum ratio is in the range of 1.8 to 1.9 for the two frequency bands.

The fourth set of data is for the N42/N43 anomaly where the main DC signal shift during the transition to the anomaly is from the N42 and N43 ex-core detectors. Again the noise levels of N42 and N43 are high, and the noise levels of N41 and N44 are low. The maximum to minimum ratio is 1.5 and 1.8 for the two frequency bands. The noise patterns for the N44/N42 and N42/N43 flow anomalies are similar, but in the former, N42 has the highest level, and for the latter, N43 has the highest level. In all cases there is a very distinct increase in the noise level unbalance between the four detectors over the no flow anomaly base line case. It appears that most of the unbalance is created by large increases in the low frequency nuclear noise levels in certain detector signals, but for the N43/N41 flow anomaly, the nuclear noise level of N43 appears to drop.

Both frequency bands appear to give the same information and thus appear redundant. The RMS magnitude of the low frequency nuclear noise is in terms of the RMS magnitude of the noise component between the specified frequency limits divided by the magnitude of the steady state or DC signal from the detector. Thus, the noise level is an RMS fractional change. The largest RMS noise magnitude is 2×10^{-3} or 0.2 percent of the DC detector signal. For sine waves, the ratio of peak to peak magnitude divided by RMS magnitude is 2.8, while for broad band noise the ratio is higher. Using a ratio of 4 gives a peak low frequency nuclear noise magnitude of about 0.8%, which is essentially the same magnitude as the anomaly DC level transition.

In the case of the N43/N41 and N44/N42 anomalies, the ex-core detectors which show the largest DC signal shift during transitions into and out of the anomaly, N43 and N44 respectively, do not show the large low frequency nuclear noise increase during the flow anomaly. In fact the N43 noise level appears to drop during the N43/N41 anomaly. For the N42/N43 anomaly, this observation is not true and the largest low frequency nuclear noise is on these two ex-core detectors.

3.6.3 In-Core Nuclear Noise at Callaway

Figure 3-21 gives the magnitude of the low frequency nuclear noise from in-core movable fission chambers located at fixed locations in the reactor core. Three xy locations, L10, L11 and E5, are monitored with the detectors near the mid-height of the core 78 inches above the bottom of the fuel. In the no anomaly condition, the low frequency nuclear noise magnitude is balanced between the three locations. In the N43/N41 anomaly condition, there is a strong unbalance with adjacent locations L10 and L11 (which are in the core quadrant adjacent to detector N44) having approximately twice the noise level as E5 (which is in the core quadrant adjacent to N43). Comparing Figure 3-21 with the N44 and N43 low frequency nuclear noise levels for the N43/N41 anomaly in Figure 3-20, demonstrates that the in-core detectors see the same direction of unbalance as the ex-core detectors except that the unbalance is slightly more pronounced. Note that the E5 location noise level also appears to drop significantly as did that from N43 during the N43/N41 anomaly. In order to properly perform spectral analysis of nuclear noise, several minutes of steady anomaly condition recording is required. At Callaway, the short duration of most of the anomalies and the movement of the incore detectors to a variety of xy and axial locations to map the magnitude of the anomaly transitions (not the noise) prevented the accurate mapping of the in-core low frequency noise levels due to insufficient record length.

Visual observation of the in-core low frequency nuclear noise data from Callaway indicates that the pattern is similar to that seen for ex-core detectors.

3.6.4 Coherence of Low Frequency Nuclear Noise

The in-core low frequency nuclear noise is highly coherent within a core quadrant and with the signal of the ex-core detector adjacent to that core quadrant. The low frequency nuclear noise from both in-core and ex-core detectors has low coherence between different core quadrants. Thus, this low frequency nuclear noise is believed to be caused by flow fluctuations or turbulence that is increased by the anomaly condition. If the low frequency nuclear noise were due to mechanical motion, a high coherence across the core would be expected (as is seen for the [ ]^{a,c} core barrel cantilever beam vibration induced nuclear noise). Thus a mechanical reason for this low frequency nuclear noise is not indicated.

3.6.5 Ex-Core Low Frequency Nuclear Noise from 4 Loop Plants Not Exhibiting Flow Anomalies (Trojan, McGuire 1 and 2, and Byron 1 and 2))

3.6.5.1 Trojan

Figure 3-22 gives the low frequency nuclear noise for the ex-core detectors at the PGE Trojan nuclear plant. This is from a nuclear noise data tape recorded in January 1980, near the end of fuel cycle 2. Both this data tape and recently obtained data from the plant do not show the flow anomaly transitions observed at Callaway and at other 4 loop plants. The nuclear noise from the ex-core detectors is quite balanced between the four channels as was seen at

Callaway for the no anomaly condition. The lack of observed transitions and the balanced low frequency nuclear noise indicate that Trojan does not have a flow anomaly condition either changing or steady state.

The top ex-core detectors, looking at the upper half of the core, have a slightly higher nuclear noise than the bottom ex-core detectors. This is consistent with the low frequency nuclear noise being due to flow fluctuations or turbulence. Flow fluctuations will cause temperature fluctuations in the coolant which will be larger at higher core elevations. These temperature fluctuations affect the local nuclear power density, and thus the ex-core detector signal.

3.6.5.2 McGuire 1 and 2

Figures 3-23 and 3-24 show the low frequency ex-core detector nuclear noise levels for McGuire Units 1 and 2. McGuire Unit 1 data was collected on October 9, 1986 and Unit 2 on October 10, 1986. Both are for the top and bottom detector signals combined and thus should be slightly higher than for the bottom detectors alone. The noise is reasonably balanced for both units with the maximum to minimum ratio of 1.3 for Unit 1 and 1.2 for Unit 2. The Unit 1 ratio is slightly higher than the no anomaly ratio of 1.2 for Callaway, but it is significantly smaller than any of the three Callaway anomaly condition ratios.

The low frequency nuclear noise levels at both McGuire units is below that from Callaway or Trojan, but the data was obtained from both units near the beginning of core design life with small moderator temperature coefficients. This is also true for McGuire 1 which has less than 20 effective full power days of core burnup. The McGuire 1 nuclear noise levels are the lowest as would be expected as this unit would have the least negative moderator temperature coefficient.

The reasonably balanced and small magnitude of the ex-core low frequency nuclear noise and the lack of observed anomaly transitions indicate that neither McGuire unit has transitional or steady state flow anomalies.

3.6.5.3 Byron Unit 1

Byron Unit 1 data was collected June 24, 1987, at 98% power. Figure 3-25 shows a time history of the four ex-core nuclear detectors and four symmetric in-core detectors in locations L5, E11, E5 and L11. The data does not show flow anomaly transitions. The noise level of the four in-core and four ex-core detector signals appears to be well balanced. Figure 2-26 gives the low frequency nuclear noise level (RMS Magnitude) for the four ex-core and the four in-core detector locations. This figure shows a well balanced low frequency nuclear noise with a maximum to minimum ratio of 1.1 to 1.2. The low frequency nuclear noise pattern resembles that of Callaway with no flow anomaly and that of Byron Unit 2, both McGuire Units and Trojan where no flow anomaly exists. The lack of anomaly transitions and the well balanced low frequency nuclear noise indicates that the anomaly does not exist at Byron Unit 1.

3.6.5.4 Byron Unit 2

Byron Unit 2 data was collected June 5, 1987, at 90% power. Figure 3-27 is a section of a strip chart recording that shows the time history of the four power range ex-core detectors N41, N42, N43 and N44 (Top + Bottom) and of four symetric in-core detector locations L5, E11, E5 and L11 (75 inches above the bottom of the core). There are no observed signal transitions like those observed at Callaway, Wolf Creek or the two Catawba Units. The reason for this slow drift in the movable detector at core location E-11 is unknown but is not characteristic of the flow anomaly. There were no flow anomaly type of transitions in the entire 2 hours and 15 minutes of tape recorded data. There is over three hours of excore data with no transitions recorded on the strip chart recorder.

The noise levels between the four ex-core detectors and between the four symetric in-core detector locations is well balanced. Figure 3-28 gives the results of spectral analysis of the data. In both frequency ranges (0.15 Hz to 0.5 Hz and 0.15 to 1.0 Hz), the noise levels are well balanced and quite low being comparable to Callaway during the no anomaly condition and to the two McGuire Units and Trojan where no flow anomoly has been observed. The maximum to minimum noise ratio ranges between 1.10 and 1.24 which is in the no flow anomaly range. It is clear that no vortex condition is present at Byron Unit 2.

3.6.6 Low Frequency Nuclear Noise from Catawba Unit 1 with Flow Anomaly

Data was collected at Catawba Unit 1 on March 31, 1987 and April 1, 1987 at 100% power. Figure 3-29 gives both the ex-core and in-core low frequency nuclear noise levels from Catawba Unit 1 which spends about [ ]b,g of the time in a flow anomaly condition. The Catawba 1 flow anomaly appears to be similar to the N44/N42 anomaly at Callaway, and the noise pattern seen in Figure 3-29 is similar to that seen for the Callaway N44/N42 anomaly shown in Figure 3-20. The four symmetric in-core detector locations, L5, E11, E5, and L11, are near the center of the core quadrants to which N41, N42, N43 and N44 respectively are adjacent. The in-core nuclear noise pattern is essentially identical to that seen for the ex-core detectors. The maximum to minimum ratio for the ex-core and in-core detector is 2.0 and 1.9 respectively which is essentially the same as seen for the N44/N42 anomaly at Callaway shown in Figure 3-20.

Figures 3-30 and 3-31 show low frequency nuclear noise data plotted on a core layout grid. This allows visualization of where the high and low noise regions are located in the core. The in-core data is positioned in the fuel assembly location where the data was obtained with movable detectors. Because of space restrictions on the figure, each in-core reading does not show a x 10^{-3} multiplier. The two figures show data from different core elevations but both are near the mid-height of the core. Figure 3-31 at 56 inches above the bottom of the fuel will have slightly lower values than Figure 3-30 at 78 inches above the bottom of the fuel. This is because a given flow fluctuation will produce a higher temperature fluctuation higher in the core.

This effect was seen in Figure 3-22 for top and bottom ex-core detectors where the top detectors had somewhat higher noise levels. The effect between Figures 3-30 and 3-31 is expected to be about 10%. The highest noise levels are observed to be in the core quadrant adjacent to ex-core detector N42 which has the highest ex-core low frequency nuclear noise level. The second highest in-core noise is in the core quadrant adjacent to N43 which has the second highest ex-core noise. The most quiet region is in the center of the core and also in the core quadrant adjacent to N44. This is the core quadrant where the dominant DC signal level shift is observed during anomaly transitions and is believed to be the position of the dominant vortex.

3.6.7 Low Frequency Nuclear Noise at Catawba Unit 2

Data was taken at Catawba Unit 2 on 5/13/87 and 5/14/87 at 100% power. Visually, the nuclear noise data appears to follow the pattern seen at other plants. Figure 3-32 is a section of strip chart showing the four ex-core detector channel signals as a function of time (which increases towards the top of the figure). At the bottom, the reactor has a N41/N43 anomaly (which is equivalent to an N44/N42 anomaly at Catawba 1 due to a different ex-core numbering system - N44 and N41 are reversed, also N42 and N43 are reversed). N41 and N44 have lower noise levels, while N42 and N43 have higher levels. N43 has the highest noise level and corresponds to N42 at Catawba 1 Which has the highest noise level in Figure 3-32.

Figure 3-33 shows the low frequency nuclear noise levels at Catawba Unit 2 both during the N41/N43 anomaly and during the no flow anomaly condition. The sequence of excore detectors has been changed from that of previous figures to account for the different ex-core locations at Catawba 2 where N41 is located where N44 is in other plants etc. The revised sequence maps the Catawba Unit 2 data into the same core reference locations used at other plants. With this in mind, the N41 anomaly plot is essentially identical to the pattern seen at Callaway and Catawba Unit 1 during their equivalent N44/N42 event. The no anomaly noise is quite balanced as is seen at Callaway for the no anomaly condition and at both McGuire units, both Byron units, and Trojan. The low frequency nuclear noise at Catawba Unit 2 is consistent with that at the other plants. The N41/N43 anomaly at Catawba 2 has a similar noise pattern to the corresponding N44/N42 anomaly at Catawba 1.

The relatively rare N42/41 event at Catawba Unit 2 did not provide sufficient record length to perform spectral analysis but Figure 3-32 shows that a strong ex-core detector nuclear noise unbalance exists during this event. At the bottom of the chart, the plant starts with an N41 anomaly present. About one third of the way up the figure, a second anomaly appears on the N42 detector signal which drops. At this time the noise level on the N43 and N44 appear to increase with N44 noise increasing significantly.

About two thirds the way up the figure, the signal on N41 increases indicating that the N41 anomaly has disappeared. At this time, the noise level on N42 and N43 drops such that all four detectors appear to have the same low noise level.

This visual inspection of the Catawba 2 strip chart segment supports the conclusion drawn from examining the low frequency nuclear noise from other plants. If the low frequency nuclear noise is balanced and low, no flow anomaly is indicated, while a significant unbalance or high levels indicate the presence of a flow anomaly.

3.6.8 Low Frequency Nuclear Noise at Indian Point Unit 2

At Indian Point Unit 2, the anomaly type transitions have been observed, but their character appears somewhat different. The transitions at Callaway, Wolf Creek, and Catawba move quite rapidly (1-3 seconds) from one state to another and remain in that state for some time before going to another state. This gives the rectangular signal shifts as seen in Figure 3-32. As seen in the strip chart recordings in Section H of the Appendix, at Indian Point Unit 2, the transitions are more gradual, perhaps due to the dome-shaped lower core support plate (vs. the flat plate lower core support plate for the other plants). Figure 3-34 shows the ex-core nuclear noise at Indian Point Unit 2 from a data tape recorded on April 20, 1983. Both the top and bottom ex-core detectors show an unbalance, although the pattern is different than any of those seen at Callaway. The dominant transitions observed at Indian Point Unit 2 are associated with the N43 channel.

Figure 3-35 gives the low frequency nuclear noise at Indian Point Unit 2 for a data tape recorded on August 14, 1986. At this time, the ex-core channel N43 was in almost continuous transition with about [ ]^{b,g} anomaly transitions in 44 minutes. The noise pattern seen on this later recording differs from the earlier data. It should be pointed out that it was not possible to separate out the periods of no anomaly from those with an anomaly as was possible at Callaway, thus the RMS levels for both Indian Point tests was averaged over the entire recording of anomaly and no anomaly conditions. Looking at the Callaway data in Figure 3-20, the N43/41 and N44/42 anomalies have opposite unbalances. If the Callaway data were averaged over the various anomaly transitions, the degree of unbalance would become significantly less, as unbalances between the N43/41 and N44/42 and the other anomalies would tend to cancel out. This appears to have happened for the two Indian Point Unit 2 recordings. In particular, the latter data set in Figure 3-35 with almost continuous transition between anomaly conditions represents a mixture of low frequency noise unbalances, some of which may cancel. Thus, in the case where there are many transitions, the low frequency nuclear noise is less useful, but the transitions themselves can be used to detect the flow anomaly. However, in the case where there are no observed transitions and a determination of whether a continuous anomaly may be present is required, this method is effective.

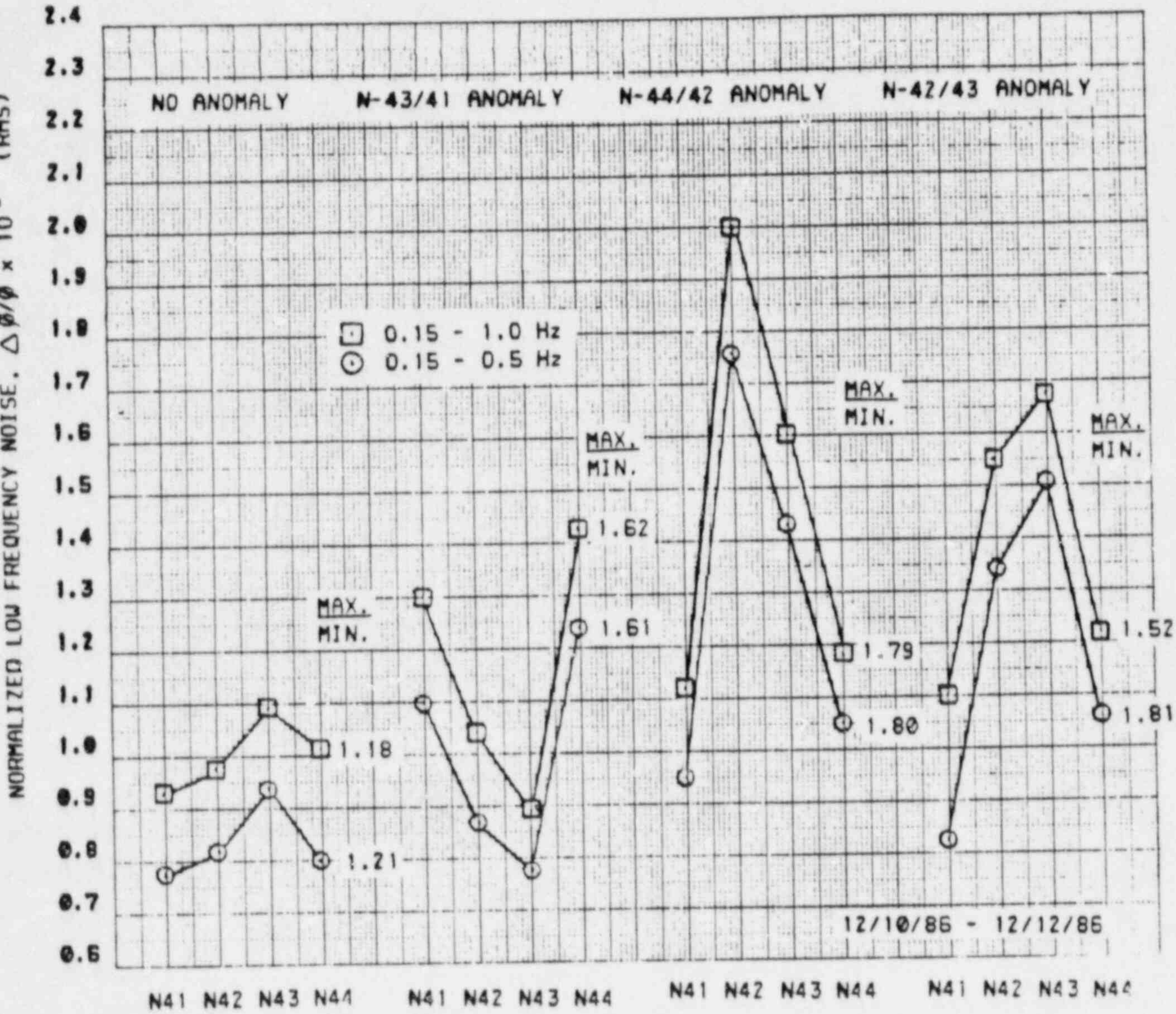

Figure 3-20 Callaway Excore Noise Levels

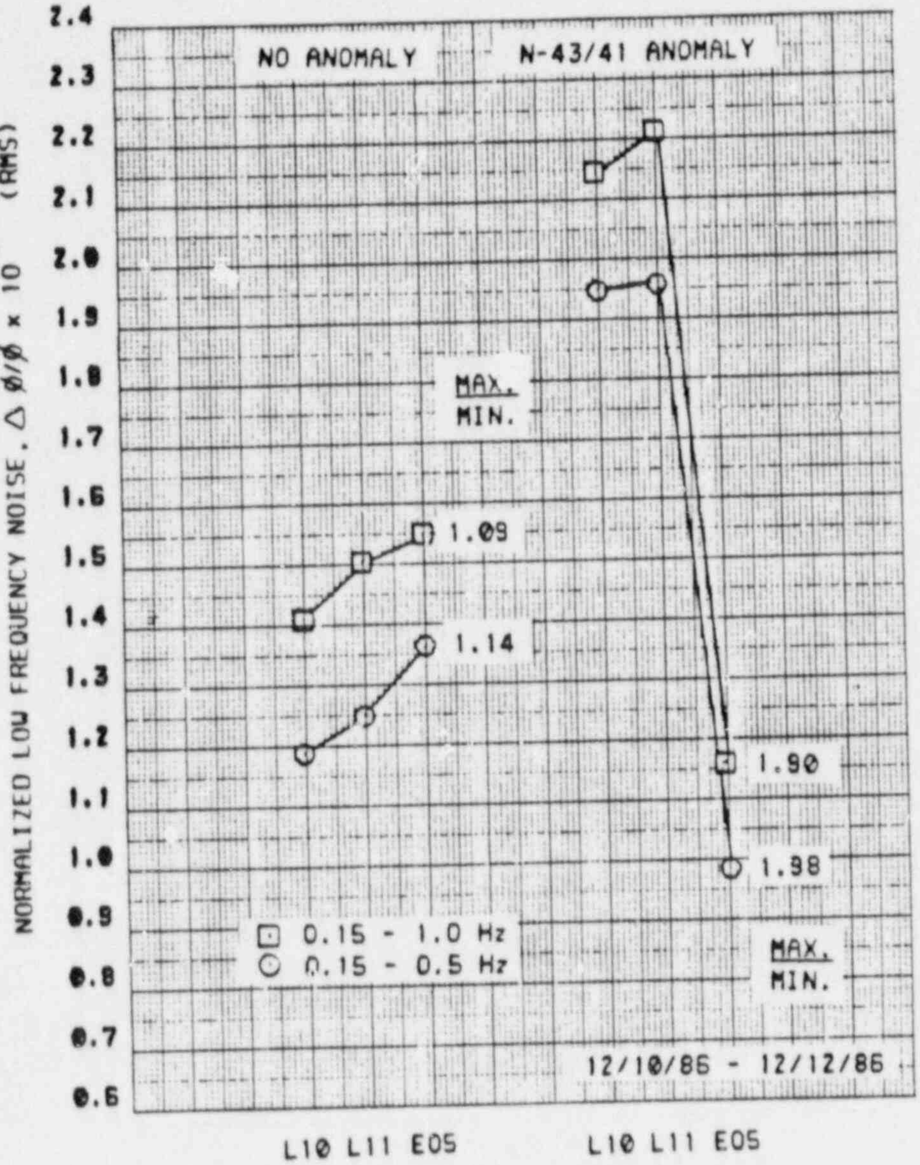

Figure 3-21 Callaway Incore Noise Levels

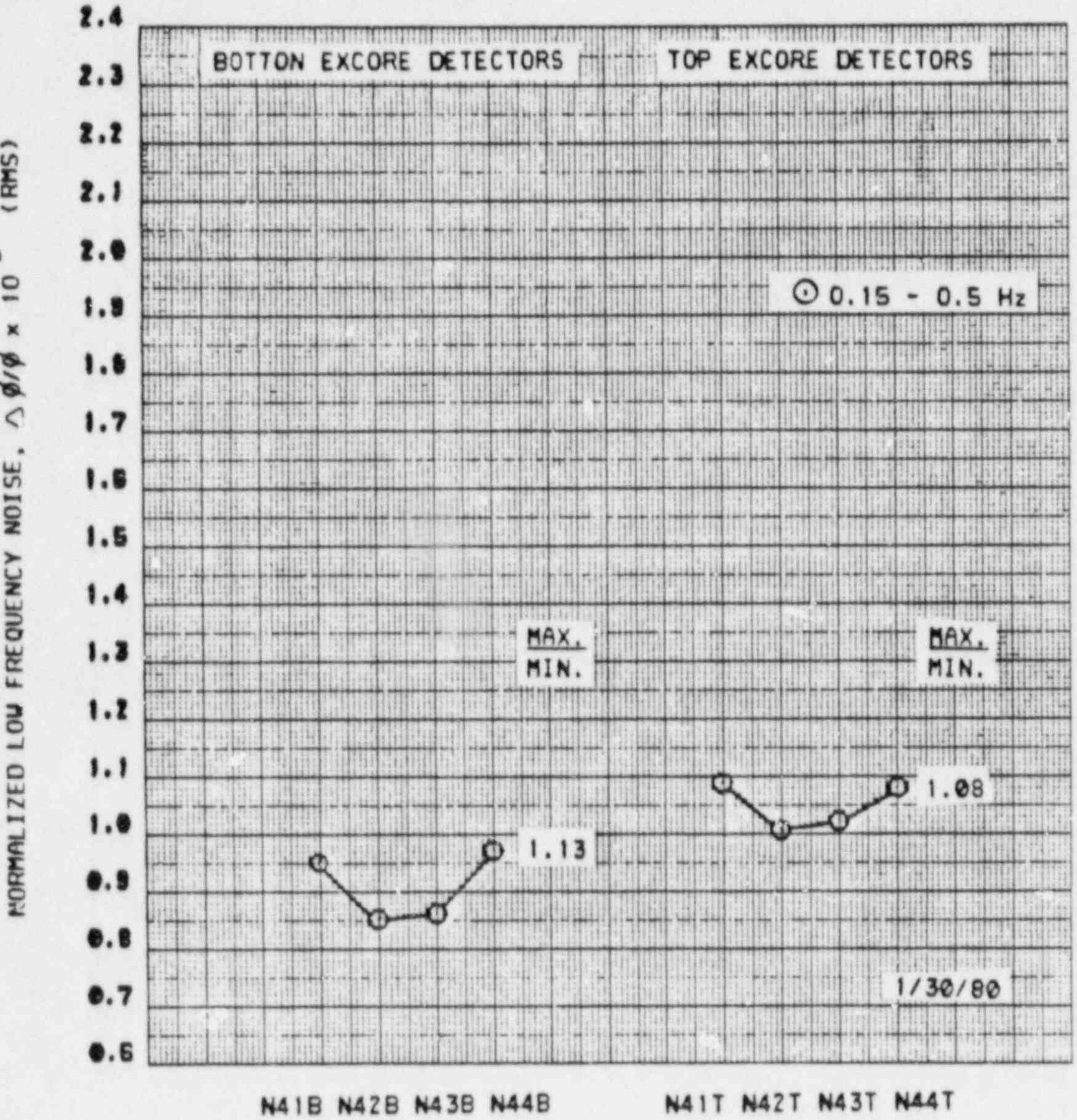

Figure 3-22 Trojan Excore Noise Levels

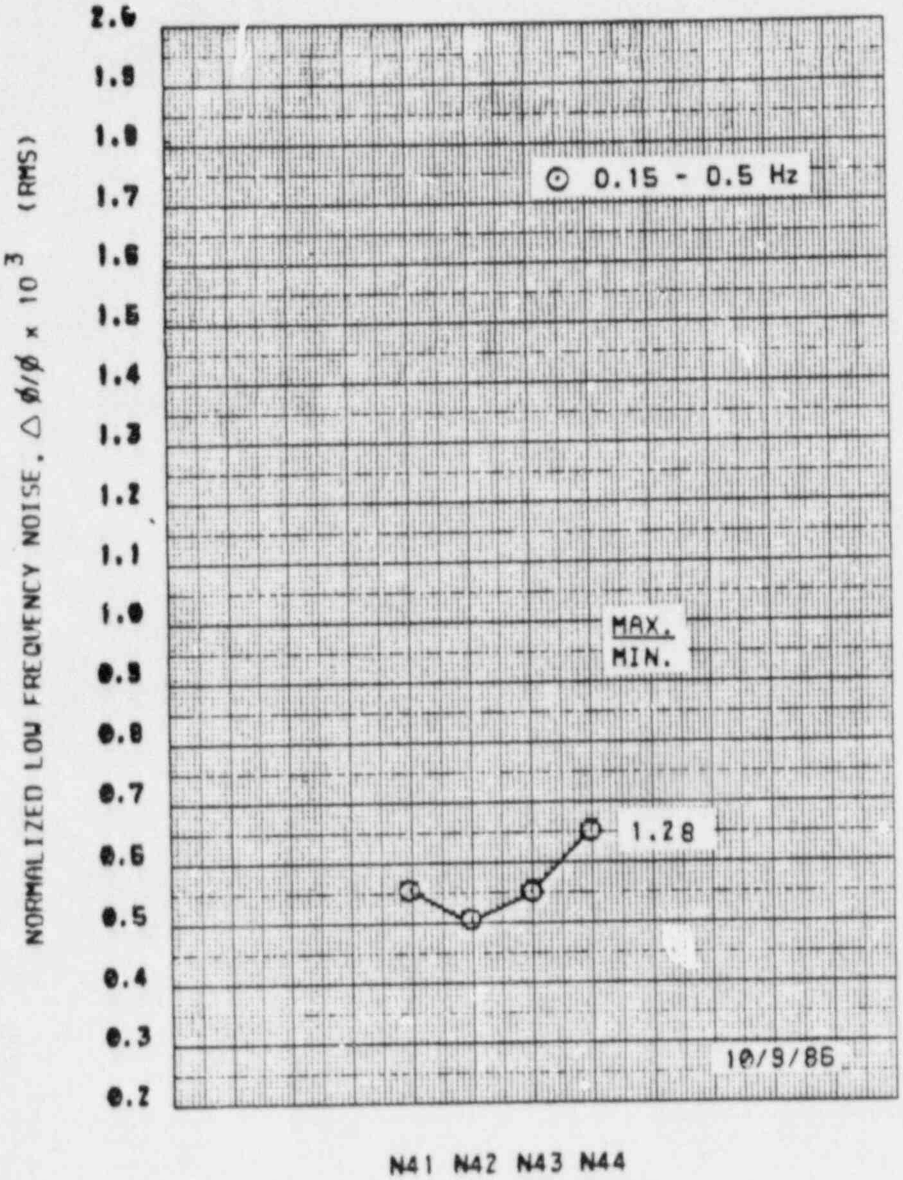

Figure 3-23 McGuire 1 Excore Noise Levels

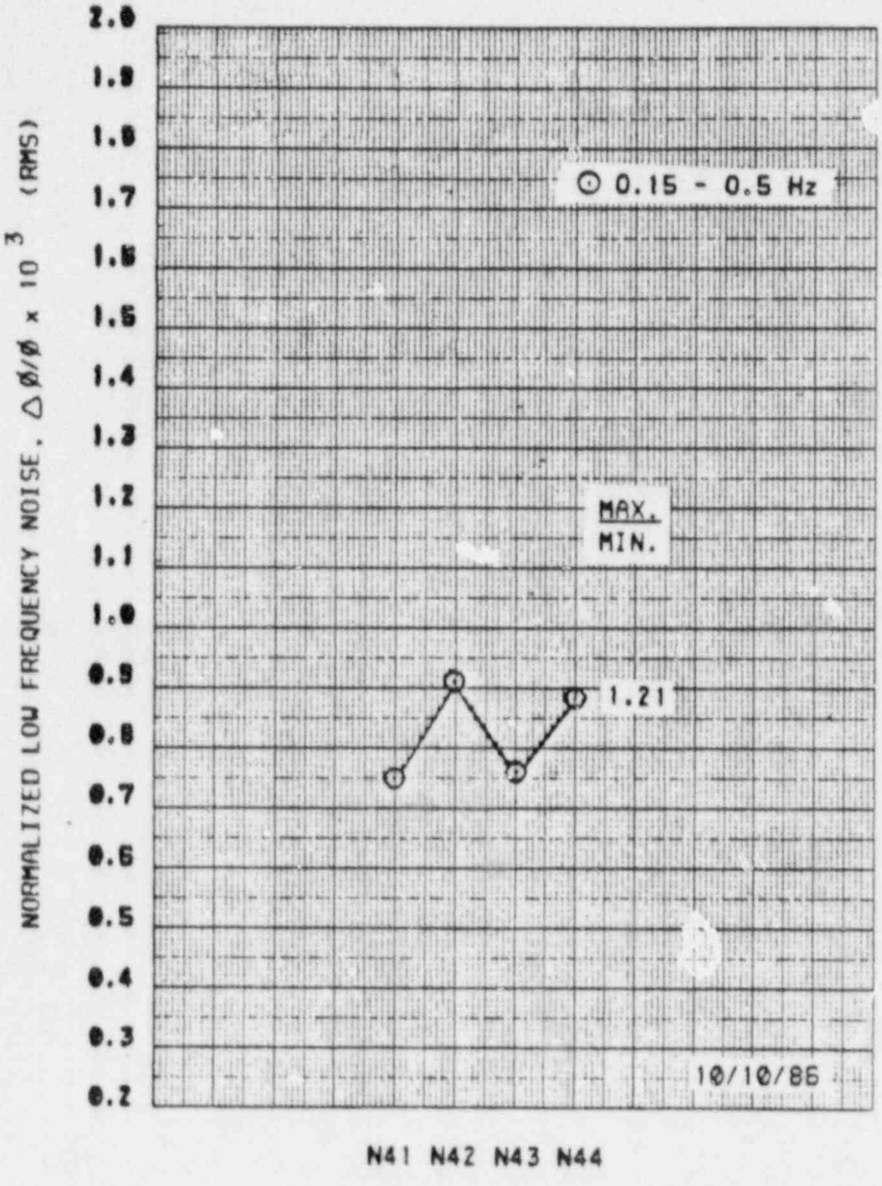

Figure 3-24 McGuire 2 Excore Noise Levels

b,g

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Figure 3-25 Byron 1 Excore and Incore Data

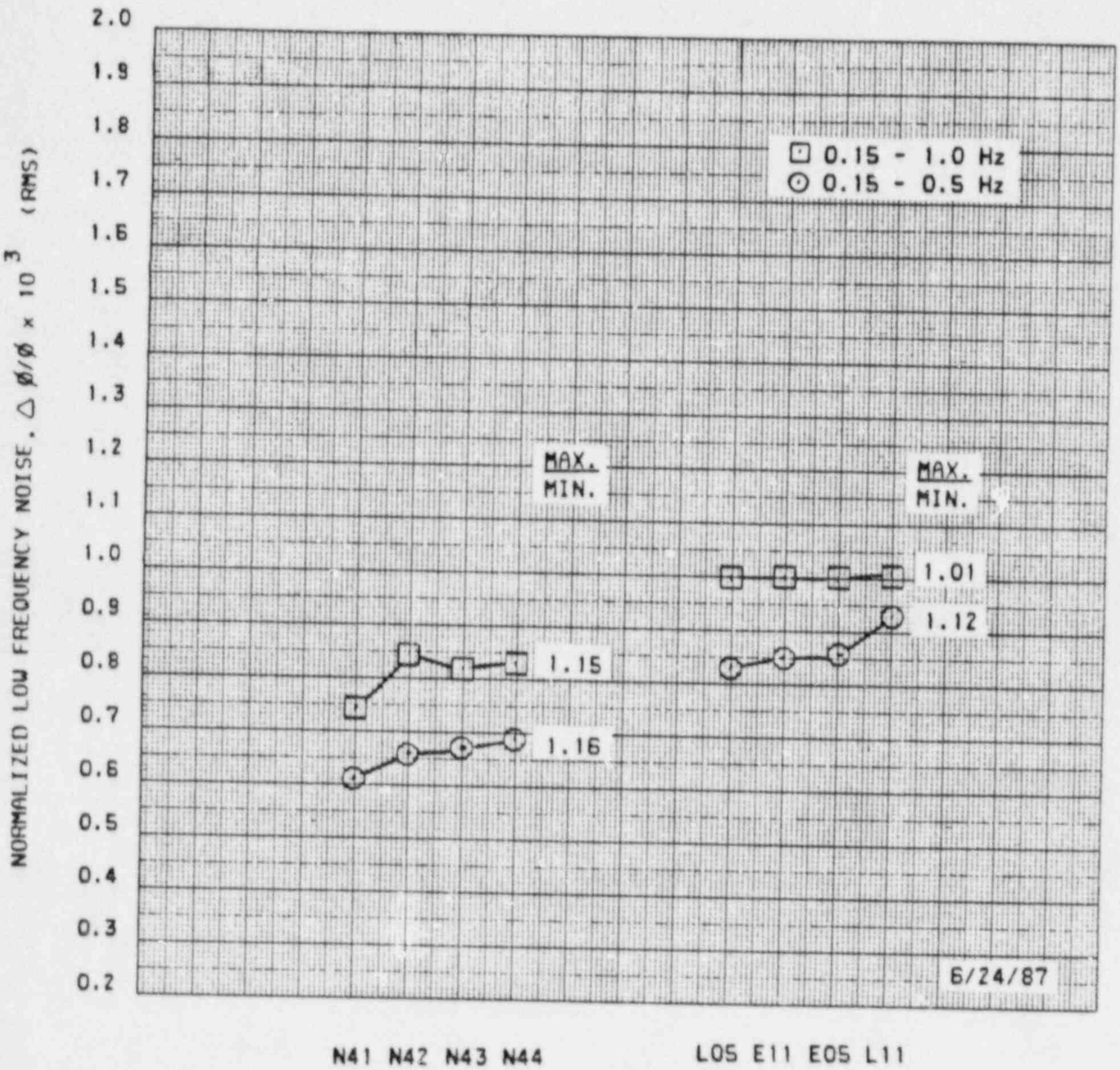

Figure 3-26 Byron 1 Excore and Incore Noise Levels

b,g

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Figure 3-27 Byron 2 Excore and Incore Data

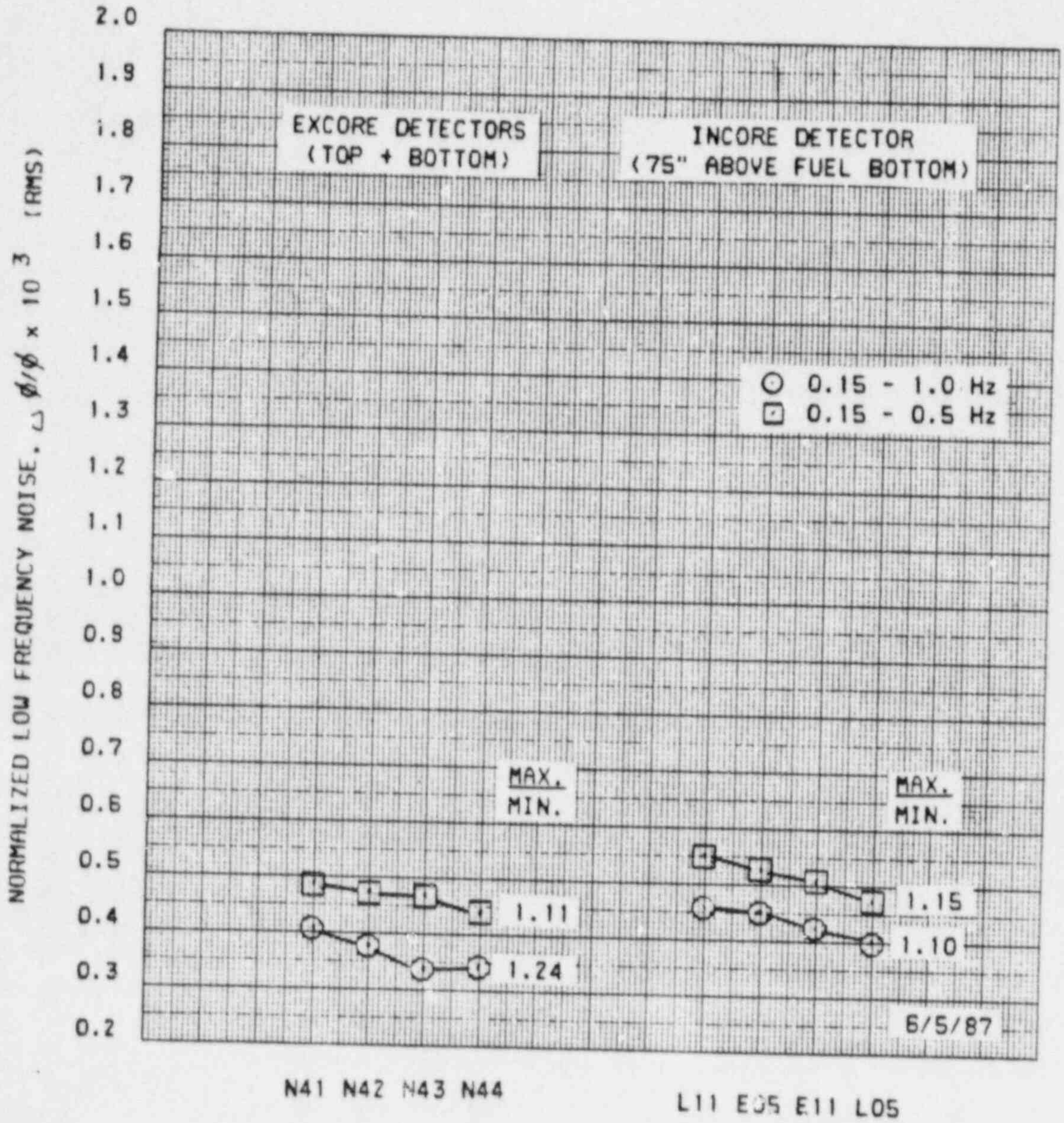

Figure 3-28 Byron 2 Excore and Incore Noise Levels

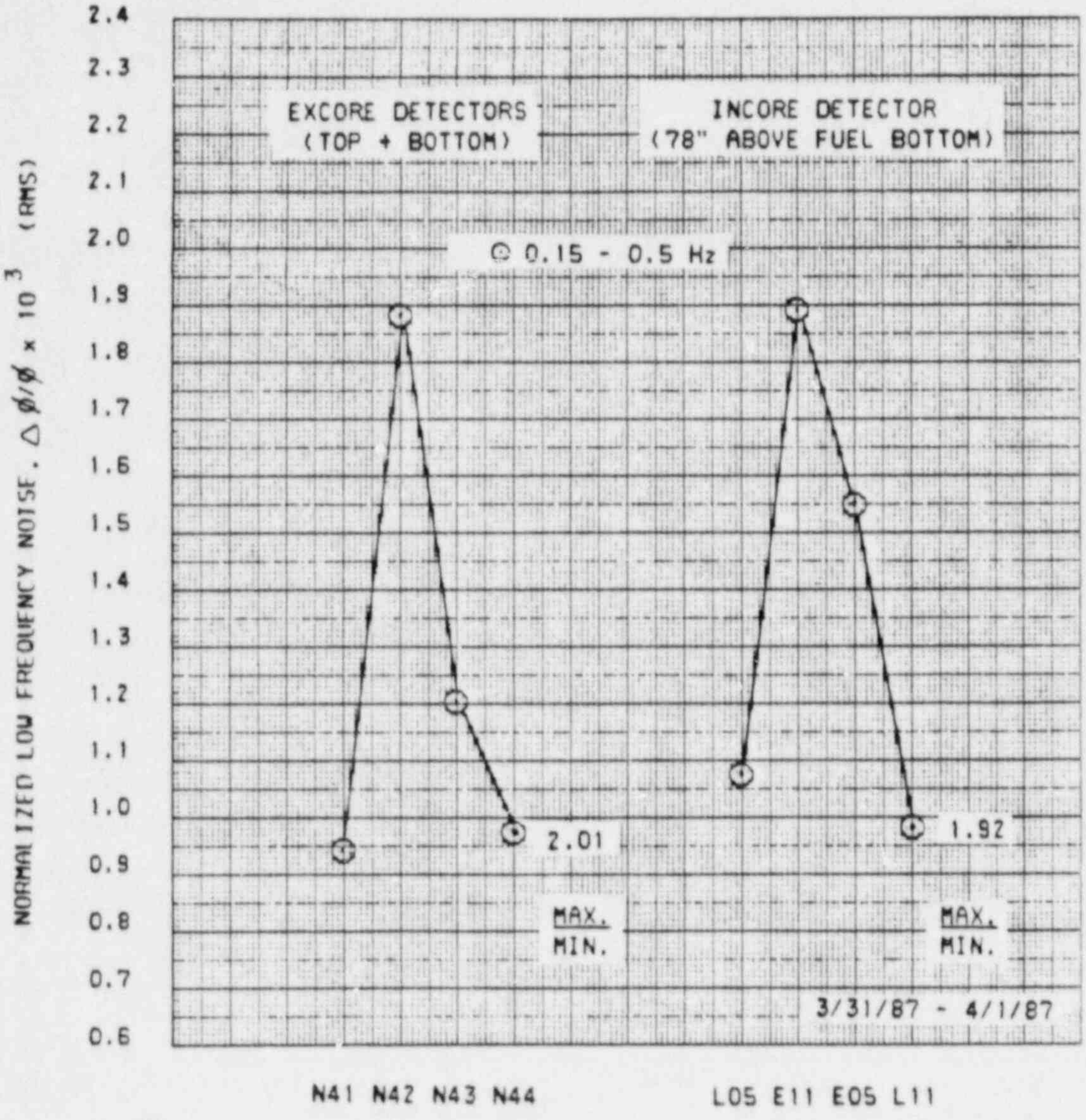

Figure 3-29 Catawba 1 Excore and Incore Noise Levels

b,g

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Figure 3-30 Catawba 1 Incore Noise Core Map

b,g

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 3-31 Catawba 1 Incore Noise Core Map

b,g

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Figure 3-32 Catawba 2 NIS and Thermocouple Data

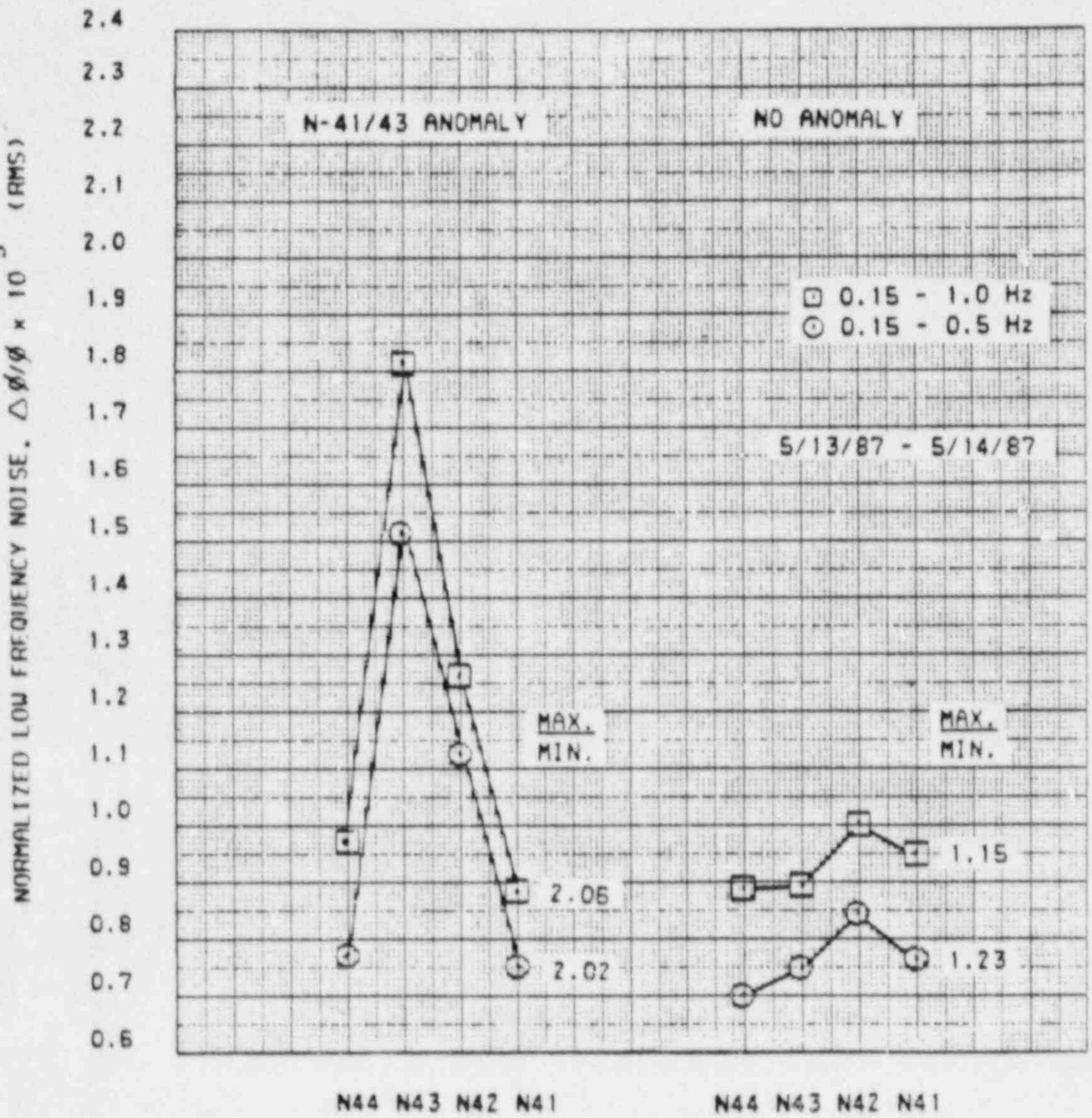

Figure 3-33 Catawba 2 Excore Noise Levels

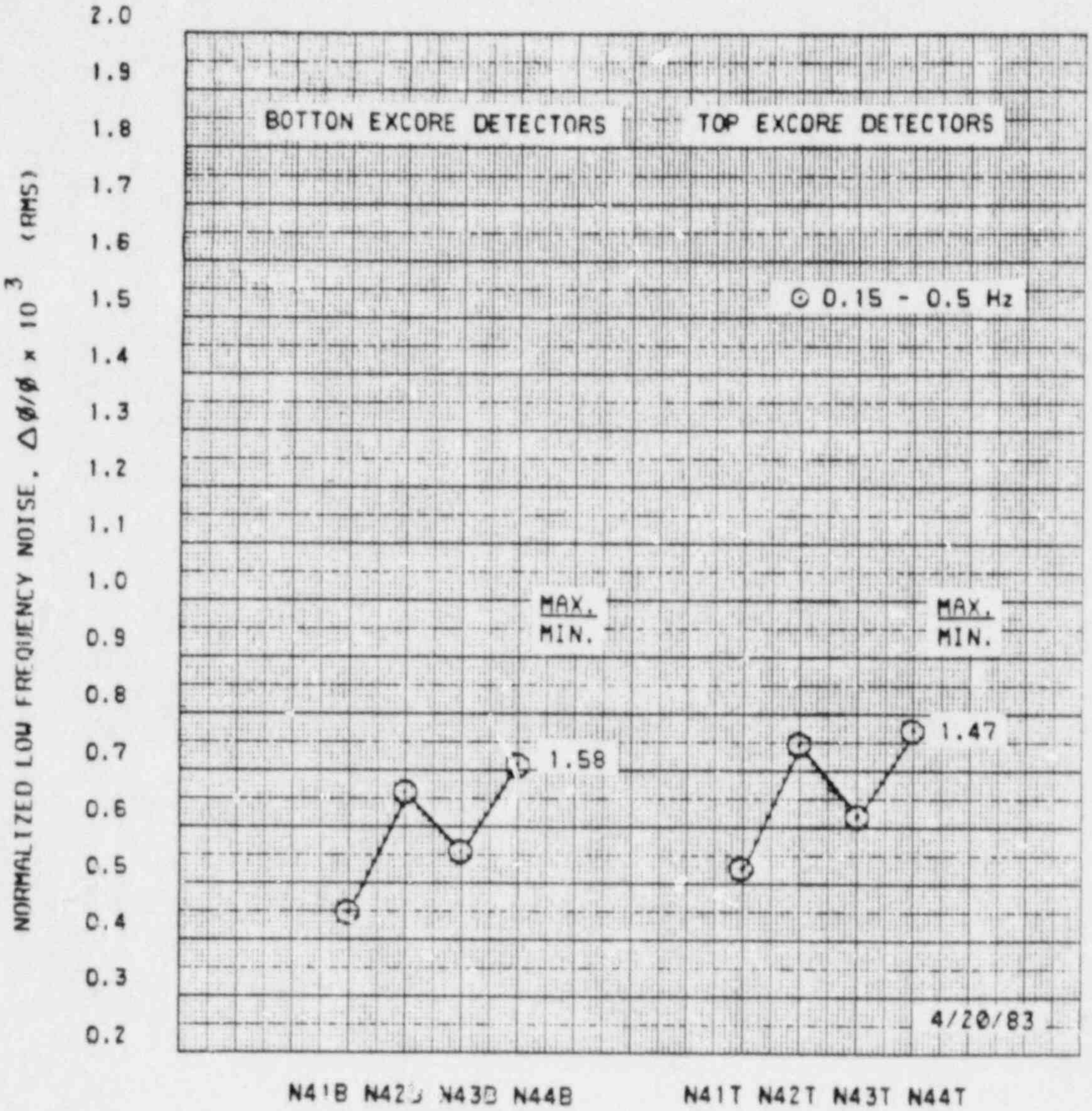

Figure 3-34 Indian Point 2 Excore Noise Levels

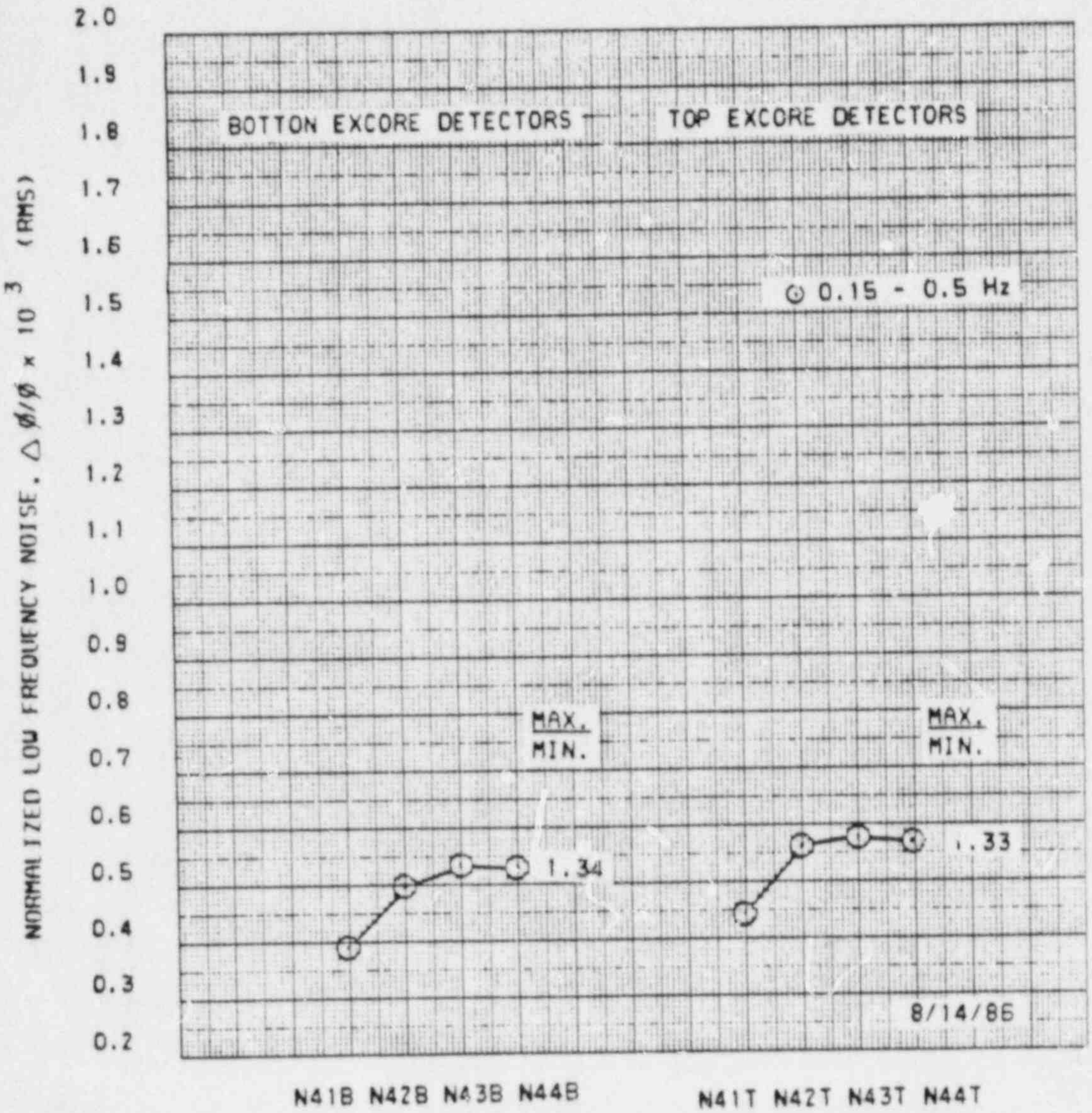

Figure 3-35 . 'an Point 2 Excore Noise Levels

4.0 SAFETY EVALUATION

The following sections describe the methodology and conclusions of the RCS Flow Anomaly safety evaluation. Cycle 2 of the Callaway plant was used as the basis for this evaluation. It is expected that the conclusions of the following sections would apply to other plants with the flow anomaly, based on a review of all flow anomaly data collected.

4.1 NON-LOCA SAFETY EVALUATION

4.1.1 Discussion

The flow anomaly may impact several items with respect to the non-LOCA safety analyses. The primary impacts are internal to the core and are caused by the inlet flow maldistribution. The discussion which follows splits the non-LOCA safety analyses into three groups. The first group is the DNBR Criteria Analyses (e.g., Uncontrolled RCCA Bank Withdrawal at Power, Excessive Load Increase, etc.). The second group is the Fuel/Cladding Temperature Transient Analyses, and the third group consists of analyses which are unaffected by the anomaly. This differentiates between the methodologies used in the analysis of these events.

4.1.2 DNBR Criteria Analyses

A summary of the impacts are as follows:

1. <u>Core Safety Limits</u> as presented in the FSAR and in the Technical Specifications. The core limits represent the limiting operating conditions with respect to Reactor Coolant System (RCS) temperature and power level and are based upon the limiting conditions of Departure from Nucleate Boiling (DNB), Core Exit Quality Limits (dependent upon the DNB correlation used), and Hot Leg Boiling. If the flow anomaly produces a flow redistribution at the core inlet which is beyond the assumption used in generating the core limits, it could impact the acceptable operating region defined by the core limits.

2. RCS Loop Flow may vary (approximately 0.3%) due to the formation and dissipation of the flow anomaly. A minimum RCS flow is assumed in the non-LOCA safety analyses. This minimum RCS flow assumption is supported by a minimum flow requirement in the Technical Specifications. If the flow anomaly were to result in flow less than the flow assumed in the safety analyses, then the flow initial conditions and core limits would be impacted.

3. RCS Loop Temperatures may vary due to the formation and dissipation of the flow anomaly. A maximum initial temperature is assumed in the non-LOCA safety analyses for all core evaluation events. This maximum RCS temperature assumption is supported by a maximum temperature limitation in the Technical Specifications. The flow anomaly at Callaway results in a slight rise in the hot leg temperature for Loop 1 and no change in the cold leg temperatures and remaining hot leg temperatures. If the flow anomaly were to result in perturbations beyond the normal temperature control band for the rod control system (typically ±1.5 [Δ°F] from the programmed T_{AVG}) then the maximum temperature assumptions in the safety analyses could be impacted.

4. The Core Peaking Factors may vary due to the flow anomaly. The Moderator Temperature Coefficient may result in power generation shifts within the core due to feedback in the anomaly affected regions.

   The range of analyzed power distributions are supported through the Technical Specifications on Axial Power Difference, Heat Flux Hot Channel Factor, Quadrant Power Tilt Ratio, and Nuclear Enthalpy Rise Hot Channel Factor monitoring requirements. Flux maps taken at Callaway determined that the measured values for assembly power, quadrant power, etc., compared well with predictions and show no significant asymmetry from quadrant to quadrant. If the flow anomaly were to result in perturbations beyond the ranges specified in the Technical Specifications, the safety analysis assumptions on initial and transient peaking factors could be affected and the core limits could be impacted.

Therefore, Callaway satisfies the Technical Specification requirements on Core Peaking Factors, RCS Flow, and RCS Loop Temperatures. For the RCS flow anomaly, an evaluation of the core limits and the limiting DNBR analysis (the Loss of Flow event) were performed.

4.1.2.1 Departure From Nucleate Boiling (DNB)

As discussed in Section 3.4, the [ ]^a model which best fits the RCS flow anomaly data has [

]^c This [ ]^a model was used with [ ]^a to determine the effect of the flow anomaly on minimum DNBR in the core. Although [ ]^a were analyzed, the [ ]^a were evaluated. In particular, analyses were performed for [ ]^a for Callaway. The [ ]^a power distribution was used. This is [

]^a and has been shown to bound all [ ]^a power shapes with respect to DNB. Also, the total reactor flow rate was decreased by [ ]^b to model the flow reduction indicated by the Callaway loop flow data during an N-44 event.

The magnitude of assembly power reductions in the perturbed region of the core depends on [

]^a

[ ]^{a} In the following sections, the DNB penalty ascribed to the flow anomaly is given for both [
]^{c}

4.1.2.1.1 [ ]^{a}

[ ]^{a} runs were made with [ ]^{a} axial power shapes; [
]^{a} These axial power shapes are shown in Figure 4-1. The selected [
]^{a} which is not permitted during normal, full power operation, but it was analyzed [
]^{a} The [ ]^{a} results are shown on Figure 4-2. For the [
]^{c}

The DNBR penalty due to the flow anomaly is [
]^{c} Re-evaluating the effect of the flow anomaly on the [
]^{c}

4.1.2.1.2 [ ]^{a}

To determine the DNBR margin impact at [
]^{a} was made. [
]^{a} The statepoint selected from the [
]^{a} was analyzed with [
]^{a}

[

][a] The resulting decrease in minimum DNBR for the reference case, assuming [

][a]

When re-evaluated using [ ][c] discussed in Section 4.1.2.1, the DNBR penalty attributed to the flow anomaly [
][c]

4.1.2.2 Conclusions - Effect of Flow Anomaly on DNB Safety Analyses

Based on the evaluation of the effect of the flow anomaly on the Callaway Cycle 2 safety analyses, it has been shown that the reduction in DNB margin is [ ][a,c] At [ ][a,c] conditions, which are representative of [
][a,c] the loss of DNB margin due to the flow anomaly is [
][a,c] At [ ][a,c]
conditions, which [
][a,c] the corresponding decrease in DNBR is [ ][a,c] When credit is taken for [
][a,c] the limiting DNBR penalty is reduced to [ ][a,c]

In Section 3.3, the largest observed perturbations were summarized and compared for the plants that were found to have the RCS flow anomaly. Only one plant, Catawba 1, was found to have a larger exit thermocouple temperature change than [ ][b] observed at Callaway and used as the basis of the safety analysis reported herein. In comparing the data from Catawba 1 with that from Callaway, it is noted that [

][b,c]

and, therefore, the Catawba 1 flow distribution is [ ][b,c]

[ ]^{b,c} analyzed based on the Callaway data. In support of this conclusion, it can be seen in Table 3-5, that [ ]^{b,c} for the Catawba 1 and Callaway Units.

It is expected that application of the [ ]^{a} model to other plants which have the flow anomaly will result in a [ ]^{c} DNB penalty similar to Callaway. The DNB penalty will vary somewhat from plant to plant due to differences in fuel type, critical heat flux correlations, and plant operating conditions. However, the [ ]^{c} penality determined in this generic evaluation [ ]^{c} the flow anomaly at plants known to be affected because [ ]^{a} It is also expected that near-term operating plants which may be affected by the RCS flow anomaly will also [ ]^{c} due to the similarity of the available data from affected plants. It is expected that the plants [ ]^{c} the reduction in minimum DNBR.

4.1.2.3 DNBR Limitations

4.1.2.3.1 Cases of Insufficient Generic Margin Available

For plants where sufficient generic DNBR margin may not be available, the change in the DNBR limitations may impact the following:

1)Overtemperature Delta-T (OTDT) Setpoint

The OTDT Setpoint is calculated so that a reactor trip will occur prior to reaching/exceeding the core limits and typically will provide simultaneous protection with the Overpower Delta-T (OPDT) Setpoint at at least one point on the core limits that is limited by the DNB limiting portion of the curve. Any movement of the DNB lines to a more limiting condition would result in a section of the core limits which may be unprotected.

2)DNBR Calculations by LOFTRAN or THINC Computer Codes

The DNBR calculations by the LOFTRAN code are based upon the core limit lines. A movement of the core limit lines to more limiting conditions would result in a non-conservatism of the analyses which depend upon the LOFTRAN calculated DNBR. An evaluation would be required to determine the impact on the LOFTRAN analyzed transients.

The DNBR calculations by the THINC code are detailed Thermal Hydraulic calculations which, in the existing analyses of record, do not account for the effects of the anomaly.

4.1.2.3.2 Additional DNBR Margin Options

In order to address these points for plants without sufficient generic margin, several potential areas may exist where margin can be gained:

1)Margin in OTDT and OPDT Setpoint Calculations

In protection system setpoint studies by Westinghouse, the amount of margin existing between the analysis assumptions on the OTDT and OPDT setpoints and the Technical Specification setpoints are identified. If margin exists in either setpoint, the reduction or elimination of the margin may provide sufficient adjustment of the core protection lines to permit coverage of the core limits. Utilization of margin existing in the High Pressurizer Pressure and Low Pressurizer Pressure reactor trip setpoints can also support this evaluation.

This option would require a verification of the revised analysis setpoint(s) with the methodology used in the protection system setpoint study and a verification of the coverage of the core limits with the revised analysis setpoint(s). This may also require revisions to the setpoint study and the core limits presented in the FSAR and Technical Specifications. A failure to provide coverage of the core limits with the revised analysis setpoint(s), using only the available margin would also require a change in the Technical Specification setpoints.

2)Margin Existing In the Existing Analysis of Record

The safety analyses typically demonstrates that a difference exists between the minimum DNBR value reached during the safety analyses and the limit DNBR for a plant's design basis. An evaluation of the plant specific penalty with respect to any existing difference may provide a sufficient argument for a Justification for Continued Operation (JCO). For limiting DNBR transients for which a sufficient difference does not exist, the alternatives listed here may be evaluated or a reanalysis of the event(s) with a reduction in the conservatism of the analysis may be required.

3)Application Of A Revised Methodology

The WRB-1 DNB correlation provides an alternate correlation with respect to DNBR evaluation than the W-3 correlation. The WRB-1 correlation was developed from Westinghouse rod bundle data and utilizes a more systematic approach in developing this correlation. With these methods, the 95/95 design criteria is satisfied at a lower minimum DNBR than the W-3 correlation. For plants which currently apply the W-3 correlation, the application of the WRB-1 correlation could offset the DNB penalty due to the presence of the flow anomaly.

The Improved Thermal Design Procedure (ITDP) accounts for variations in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters to statistically obtain a DNB uncertainty factor. ITDP has demonstrated margin with respect to the Standard Design Procedure. If this method has not already been applied to a plant's safety analysis, the application of ITDP may provide margin which could offset the DNB penalty due to the presence of the flow anomaly.

Additional margin may be obtained by using the Revised Thermal Design Procedure (RTDP). This methodology removes some of the conservatism in the ITDP by statistically combining the system and DNB correlation uncertainties. With ITDP, the system uncertainties are statistically combined separately from the DNB correlation uncertainty and, then, the two are combined directly rather than statistically, to determine the DNBR limit. A topical report on RTDP was submitted to the NRC for approval in April 1987.

4.Measured RCS Flow

If the measured RCS flow at a plant is greater than the minimum flow requirement of the Technical Specifications, the additional flow can be used in an evaluation to offset the DNB penalty due to the presence of the flow anomaly.

Transients which are included in this category of "DNBR Criteria Analyses" are:

-Loss of Load/Turbine Trip
-Uncontrolled RCCA Bank Withdrawal at power
-Feedwater Malfunction
-Excessive Load Increase
-Accidental RCS Depressurization
-Inadvertent ECCS Initiation
-RCCA Misalignment (Dropped Rod)
-Steamline Break (At Power, Zero Power)
-Loss of Flow/Frequency Decay
-Boron Dilution (Modes 1+2)
-Startup of an Inactive Loop

4.1.3 Fuel/Cladding Temperature Transient Analyses

The impacts of the flow anomaly are as follows:

1. The anomaly results in slightly higher temperatures in the affected core channels. This results in a slightly degraded capability to remove heat from the fuel and clad in these channels.

2. The anomaly results in lower flow rates in the lower regions of the channel. Based upon the effects of increased cross flow in the channels, this effect is significantly reduced within the first few feet of travel in the core.

For the Callaway Cycle 2 evaluation, two transients were reviewed with respect to the fuel/cladding temperature transients, The Rod Ejection Analyses (at full power) and the Locked Rotor analysis. Since the methodology in both analyses assume that a DNB condition occurs immediately at the initiation of the transient, the heat transfer coefficient used is relatively insensitive to the coolant bulk temperature and is only slightly sensitive to the flow rate in the channel. The more limiting transient is the Rod Ejection at Power. A review of the full power rod ejection events was performed to identify the transient peak power limitations in the lower regions of the core (approximately the first three feet). Maximum ejected rod worths are obtained from initial flux distributions which are peaked toward the top of the core. These maximum ejected rod worths are used in the FSAR analysis for the rod ejection events at full power. [

][a,c] As a result, the effect of only the average flow variation [ ][b,g] on the FSAR analysis is required in order to bound the effects of the flow anomaly on the FSAR. Based upon available sensitivities, the [ ][b,g] flow variation results in a negligible impact on the results in the FSAR analysis and the Callaway analysis demonstrates more than sufficient margin to accept this perturbation. The locked rotor event provides even more margin to the temperature limits.

4.1.4 Unaffected Analyses

The following analyses were evaluated to not be affected due to the fact that they are analyzed with only two pumps in operation, where an anomaly condition has not been indicated to be present based upon the test data taken at Wolf Creek:

RCCA Withdrawal From Subcritical
RCCA Ejection (Zero Power)

Some analyses depend upon the overall response of the core in the form of its power/reactivity transient since these analyses evaluate criteria which are external to the core, (e.g., containment temperature and pressure response, decay heat removal capability, peak pressure reached during the transient, etc.) The following analyses were evaluated to not be affected due to the fact that they are analyzed for criteria which only depend upon the gross response of the core and RCS:

Loss of Normal Feedwater/Station Blackout
Feedline Break
Steamline Break Mass/Energy Release
Boron Dilution (Modes 3-6)

b

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 4-1 Axial Power Shapes Used for
[ ]^{a} Analyses

b,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 4-2 DNBR vs Elevation During [ ][a] Transient

4.2 LOSS OF COOLANT ACCIDENT SAFETY EVALUATION

This evaluation covers the large break and small break LOCA issues which are associated with the RCS Flow Anomaly as first reported at Callaway. This evaluation is predicated upon a mechanism which is consistent with the observations and data analysis. In particular, a rotational flow cell or cells in the lower plenum in the reactor vessel is assumed to produce a local reduction in core inlet flow during normal operation.

In addressing the limiting large break LOCA (LBLOCA), it is assumed that the break is initiated during an event and that the initial core hydraulic conditions and fuel rod temperatures have been determined by the THINC analysis, Section 3.4. Since the THINC analysis has shown margin to DNB, the fuel rod hot spot temperature is not expected to be impacted by the event and begins the LOCA transient at essentially the same values as in a non-event condition. With the source of the hydraulic disturbance being a rotational cell(s) in the lower plenum and with the limiting LB LOCA being a double ended break in the cold leg, the hydraulic effects of the event will be minimal. For the cold leg break, the initial flow in the core is upward. However, as shown in THINC calculations, the flow reduction at the core inlet persists to approximately [ ]^{a,c} the core height and so that the normal flow rate has been established at the core hot spot, i.e. at the core mid-plane. During the upflow portion of the transient, no reduction in hot spot cooling occurs. After ten to twelve seconds of upflow, core flow reversal occurs and results in the core cooling flow coming from the upper plenum. Since the event has been observed to have a minor impact on core outlet conditions, the initial flow conditions in the upper plenum are identical with or without the event and the cooling flow to the core is not impacted. For the refill and reflood periods, the vessel is essentially filling and water is slowly flowing into the core. The core reflood velocities are sufficiently low, 1 to 2 inch per second, that ample flow redistribution can occur at the bottom of the core. The flow at the quench front is uniform and no change occurs in the core heat transfer. Thus, the effect of the event on the limiting large break LOCA is believed to be insignificant.

In addressing the small break LOCA, it is recognized that the significant period of rod heatup is not initiated until core uncovery, long after loop flow and lower plenum flow have stopped. Thus, the effect of a hydraulic disturbance in the lower plenum is expected to dissipated well before rod heatup is initiated and the condition is not expected to have any impact on small break peak clad temperatures.

4.3 STRUCTURAL EVALUATION

The flow anomaly investigation included an assessment of the structural margins of safety in affected reactor internal components. In order to determine the effect on the existing margins of safety it was necessary to (1), identify the types of plants having the anomaly, and (2), the reactor internal components most affected by the anomaly. From a review of plant data, it appears that certain types of 4-loop plants are more susceptible to the anomaly than either 2-loop, or 3-loop plants. Furthermore, Westinghouse's examination and correlation of in-core and ex-core detector measurements as well as core outlet temperatures, elbow tap measurements and RVLIS detector pressure measurements has led to the conclusion that the reactor internals component most subjected to the velocity, flow and pressure perturbations caused by the anomaly are the bottom mounted instrumentation support columns, tie plates, and bolts.

The bottom mounted instrumentation components are classified as internal structures and not core support structures - in accordance with the ASME Code. Consequently, the concern with these components is more of a functional concern rather than a structural one. However, Westinghouse has performed a generic structural assessment on these components and concludes that there is sufficient structural and fatigue margin, so that no loss of function is expected to occur for even higher flows and velocities than currently determined for the most limiting 4-loop plant.

The dynamic response of the 17x17 standard and OFA fuel rods were analyzed to determine their sensitivity to the flow anomaly. The fuel rod natural frequencies and mode shapes were determined using a finite element code (WECAN). The mode shapes are predominantly determined by [

]c The natural frequencies are a function of the bending stiffness and mass. [

]c The mode shapes and natural frequencies were combined with the crossflow velocities due to the flow anomaly to determine the fuel rod response sensitivity. Crossflow velocities for use in the fuel rod dynamic analyses were obtained from [

]a. The axial profiles of the crossflow velocity is shown in Figure 4-3. Dynamic analyses made with this crossflow velocity distribution show that [

]c

In conclusion, [

]c

a,c

THIS FIGURE PROPRIETARY IN ITS ENTIRETY

Figure 4-3 Axial Distribution of Maximum Crossflow Velocity

5.0 REFERENCES

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4. "Hydraulic Studies of the Lower Plenum of a Pressure Water Reactor," Siemens Corp., NUREG-tr-0009, Published by NRC: October 1977.

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APPENDIX - PLANT DATA

This appendix records the plant data made available for the flow anomaly investigation. The investigation began at the Callaway plant November 1986, where signals were recorded on strip charts. Multiple signals could be recorded simultaneously and changes seen clearly from these recordings. The Wolf Creek data was recorded likewise. Subsequent data was recorded for the most part with a data logging system and transmitted as plots or print out of the digitized data.

Data received in digital form was processed with a PC spreadsheet and data averaging used to make signal changes more discernable. Plots of data processed in this manner are presented.

The data is presented in the following order.

A. Callaway
B. Wolf Creek
C. Catawba Units 1 and 2
D. McGuire Units 1 and 2
E. Millstone 3
F. Diablo Canyon Units 1 and 2
G. Trojan
H. Indian Point 2
I. Indian Point 3
J. Beaver Valley
K. Shearon Harris
L. Prairie Island Unit 2
M. Ginna
N. South Texas Unit 1
O. Byron Units 1 and 2

A. Callaway

The first data recorded in this investigation was at the Callaway plant, November 1987. Plant signals were taken on continous recording strip charts, including excore neutron detectors, in core exit thermocouples, primary loop flow, RVLIS, and some secondary side parameters.

Figure A1 shows a time frame typical of the N44 and N42/43 event. Excore detector, thermocouple, and RVLIS response is shown.

Figure A2 shows a time frame typical of the N41 event. Excore detector, thermocouple, and RVLIS response is shown.

Figure A3 - A5 show the thermocouple and RVLIS response to N44 events, and recordings of main coolant pump parameters.

During the investigation site personnel recalled that excore signals had been investigated near the end of Cycle 1. Strip charts recorded at the time were retrieved and responses similar to those in Cycle 2 were observed.

B. Wolf Creek

The second plant to record data relative to this investigation was Wolf Creek. The plant returned from a refueling outage in December 1986, and recorded data on strip charts similar to that recorded at Callaway in early January 1987. Examples of parameter responses are included.

C. Catawba Units 1 and 2

The Catawba data for both units was acquired with a data logger and plots supplied. All of the data from both units is included. Subsequently, data was collected on magnetic tape. Selected examples are included.

D. McGuire Units 1 and 2

The McGuire data was acquired and reported as for Catawba. Data from selected time frames is included in this report. Shown is 40 minutes of data for each unit. A total of 2 hours and 40 minutes of data was acquired on each unit.

E. Millstone 3

Aperture cards of data recorded on strip charts was received. These were enlarged and printed. Data was recorded for a total of four hours. A selected time frame is shown here. Figures E-1 and E-2 show the response of the excore detectors and loop flows. Figures E-3 and E-4 show the response of 8 incore thermocouples.

F. Diablo Canyon Units 1 and 2

Data was received from Diablo Canyon Units 1 and 2 as logged with a system which accessed signals every 0.2 second. This data base was then processed digitally to produce plots which were transmitted. These plots are attached in their entirety.

G. Trojan

Trojan was acquired March 31, 1987 with a data logger and received in the form of printout. The data was processed with a PC spread sheet, and averaged to aid in the identification of signal response. Selected plots of the data are included.

In addition, data was available from previous neutron noise measurements utilizing excore detectors. Examples of these traces are included, which further supports the more recent data.

H. Indian Point 2

Indian Point 2 data was received in two forms. The first set consisted of printout taken January 19, 1987 from a data logger. This was processed with a PC spreadsheet program and plotted. These are depicted in the first three figures.

Subsequently, the site repeated data acquisition and supplied plots of this data. Selected plots are attached.

In addition, data was available from previous neutron noise measurements utilizing excore detectors. Examples of these traces are included, which further supports the more recent data and indicates the phenomena has been present for years.

I. Indian Point 3

Indian Point 3 data was acquired with a data logger on April 23, 1987 and transmitted by floppy disk, with the exception of thermocouples, which were logged with a multipoint recorder. The data was processed with a PC spreadsheet program. Selected plots are included. The multipoint chart showing thermocouples was not suitable for reproduction. A total time of 2 hours was logged. Plots over a 40 minute duration are included.

J. Beaver Valley

Beaver Valley was received as printout from a data logger taken February 26, 1987 and processed using a PC spreadsheet program. Plots of the processed data are included.

K. Shearon Harris

Data received to date consists of RVLIS response at 30% power. It was acquired with the plant computer and plots transmitted. Plots received are included.

L. Prairie Island Unit 2

Data was supplied in the form of plots from a data logger. The plots are included in their entirety, covering a time duration of 35 minutes.

M. Ginna

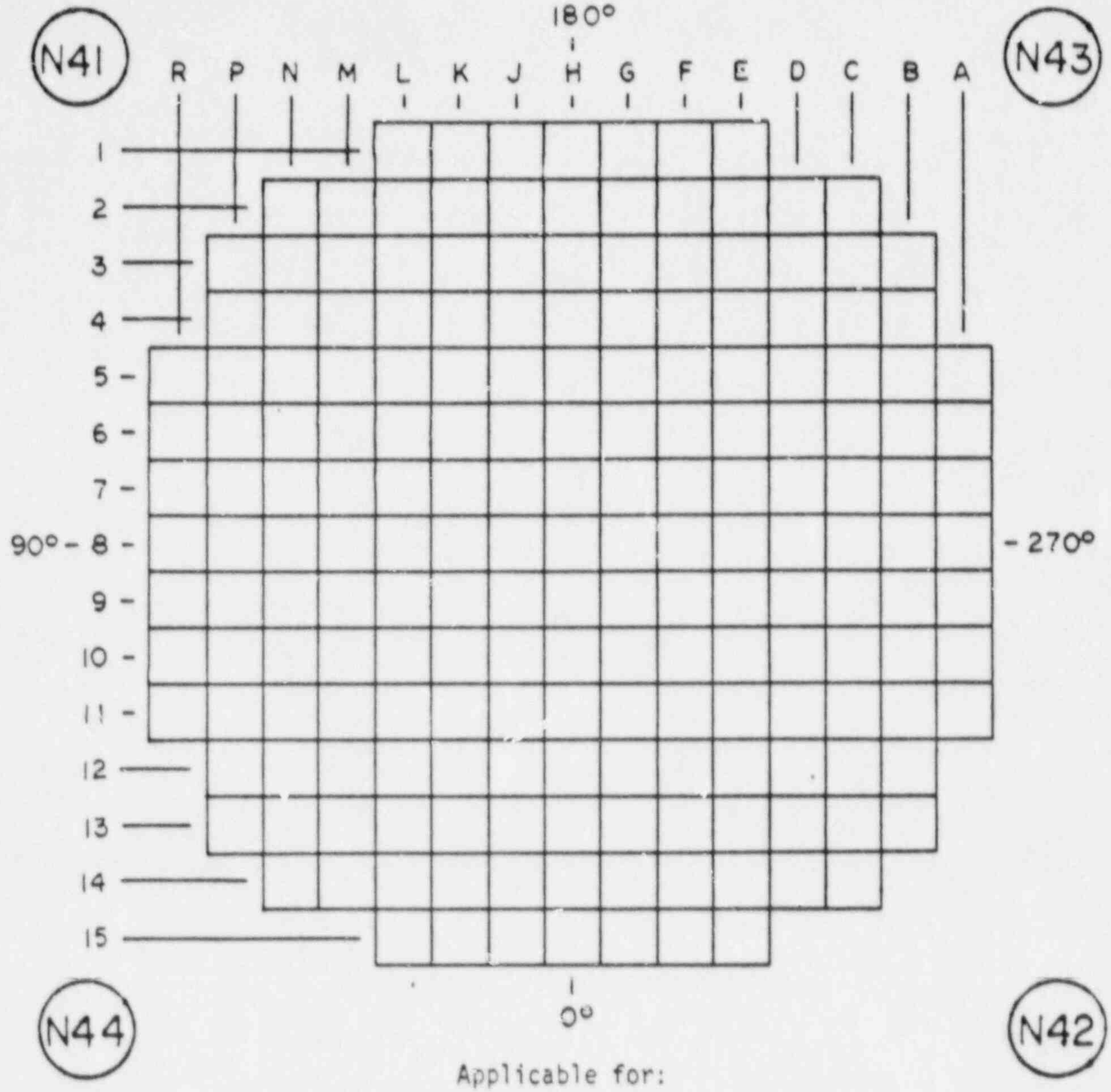
Ginna data was received as printout from a data logger taken March 31, 1987. The data was processed with a PC spreadsheet, and plotted. Data was recorded at 5 second intervals. Selected plots of the signals are included.

N. South Texas Unit 1

During hot functional testing at South Texas Unit 1, recordings were made of loop flow on February 22, 1987. A sample of these recordings is included.

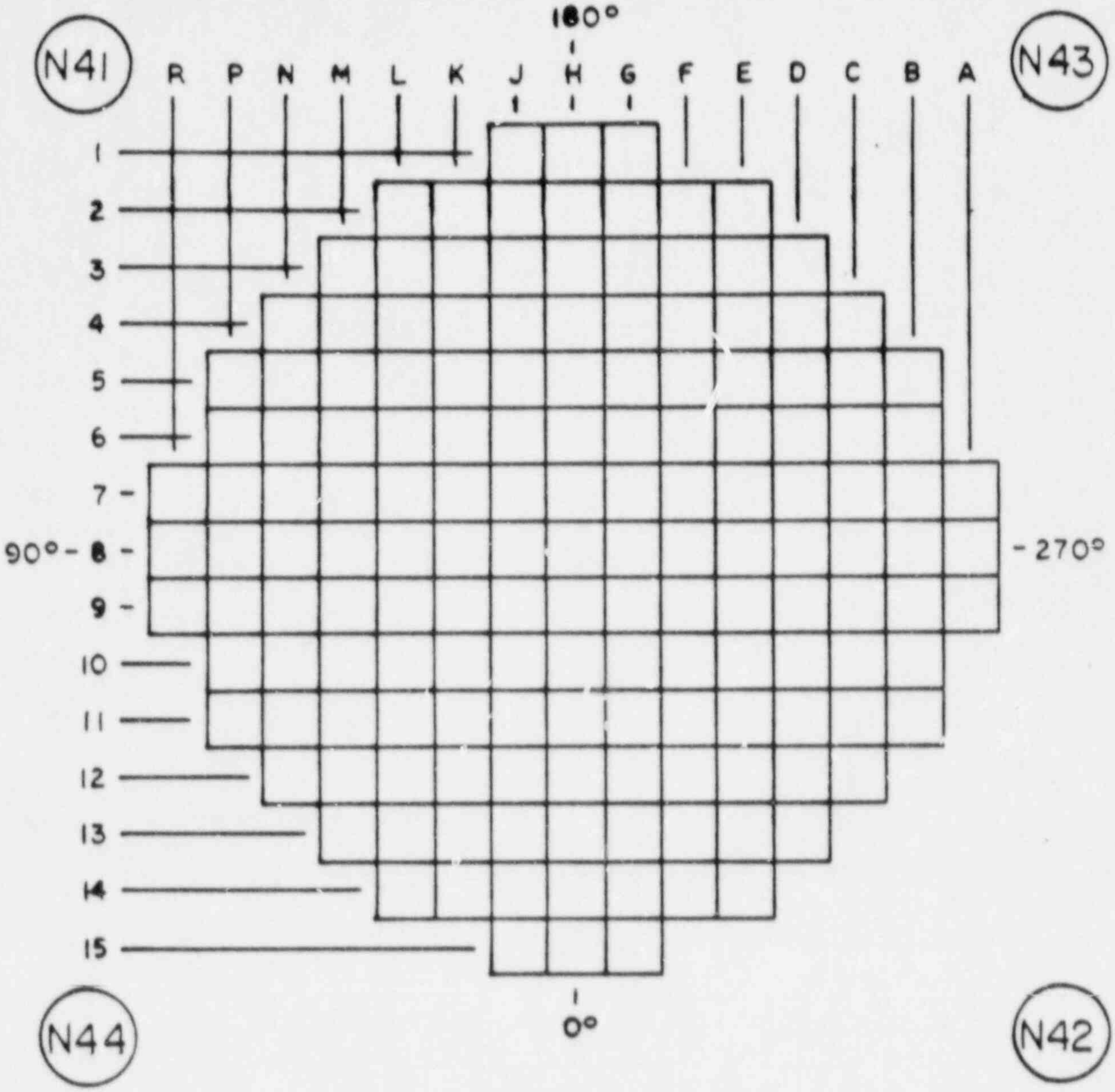
O. Byron Units 1 and 2

Data was obtained for both units with a tape recorder and the signals later played back on strip charts. No aperiodic signals of the type being investigated were observed. Selected data for both units is included.



Applicable for:

Callaway	Wolf Creek
Catawba Unit 1	McGuire Units 1 & 2
Millstone 3	Diablo Canyon Units 1 & 2
Trojan	Indian Point 2 & 3
South Texas Unit 1	Byron 1



Applicable for:

Shearon Harris
Beaver Valley 1

a,c

CALLAWAY SPECIFIC DATA ON PAGES 128 - 145 IS PROPRIETARY IN ITS ENTIRETY.

a,c

WOLF CREEK DATA ON PAGES 147 - 157 IS PROPRIETARY IN ITS ENTIRETY.

a,c

CATAWBA UNITS 1 & 2 SPECIFIC DATA ON PAGES 159 - 196 IS PROPRIETARY IN ITS ENTIRETY.

a,c

MCGUIRE UNITS 1 & 2 SPECIFIC DATA ON PAGES 198 - 218 IS PROPRIETARY IN ITS ENTIRETY.

a,c

MILLSTONE UNITS 3 SPECIFIC DATA ON PAGES 220-223 IS PROPRIETARY IN ITS ENTIRETY.

a,c

DIABLO CANYON UNITS 1 & 2 SPECIFIC DATA ON PAGES 225 - 240 IS PROPRIETARY IN ITS ENTIRETY.

a,c

TROJAN SPECIFIC DATA ON PAGES 242 - 249 IS PROPRIETARY IN ITS ENTIRETY.

a,c

INDIAN POINT UNIT 2 SPECIFIC DATA ON PAGES 251 - 264 IS PROPRIETARY IN ITS ENTIRETY.

a,c

INDIAN POINT UNIT 3 SPECIFIC DATA ON PAGES 266 - 273 IS PROPRIETARY IN ITS ENTIRETY.

a,c

BEAVER VALLEY UNIT 1 SPECIFIC DATA ON PAGES 275 - 276 IS PROPRIETARY IN ITS ENTIRETY.

a,c

SHEARON HARRIS SPECIFIC DATA ON PAGES 278 - 282 IS PROPRIETARY IN ITS ENTIRETY.

a,c

PRAIRIE ISLAND UNIT 2 SPECIFIC DATA ON PAGES 284 - 287 IS PROPRIETARY IN ITS ENTIRETY.

a,c

GINNA SPECIFIC DATA ON PAGES 289 - 296 IS PROPRIETARY IN ITS ENTIRETY.

a,c

SOUTH TEXAS UNIT 1 SPECIFIC DATA ON PAGES 298 - 299 IS PROPRIETARY IN ITS ENTIRETY.

a,c

BYRON UNITS 1 & 2 SPECIFIC DATA ON PAGES 301 - 304 IS PROPRIETARY IN ITS ENTIRETY.