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LIC-88-823

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

References: 1. Docket No. 50-285

2. NRC Safety Evaluation from NRC (P. D. Milano) to OPPD (K. J. Morris) dated July 14, 1988

Gentlemen:

SUBJECT: Clarification of Information Relating to the Operation of the Internals Vibration Monitoring System

The purpose of this submittal from the Omaha Public Power District (OPPD) is to provide clarification of information relating to the operation of the Internals Vibration Monitoring (IVM) system at the Fort Calhoun Station. The information was requested in the July 14, 1988 Safety Evaluation relating to the threshold levels for the IVM system (Reference 2). The attached report is the formal documentation of the discussion held during a telephone conversation on June 27, 1988 between Mr. J. Fisicaro and other OPPD staff members and Messrs. P. Milano and L. Lois of the NRC.

It is OPPD's belief that the proposed thresholds for the IVM system are valid based on an adequate inspection of the reactor internals and the evaluation of experiences at similar Combustion Engineering facilities, and that OPPD's staff has sufficient expertise to use the IVM system.

The attached report provides supporting documentation to resolve concerns raised in the SER and the June 27, 1988 telephone conversation. OPPD believes this information, in conjunction with previous OPPD submittals relating to the thermal shield deferral, addresses the concerns raised in the Safety Evaluation.

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OPPD believes, along with Combustion Engineering, that by using the proposed threshold levels OPPD is able to detect early signs of degradation of the thermal shield support mechanism. It is requested that the NRC perform a reevaluation of the proposed threshold levels using the attached report for supporting information.

If you have further questions concerning this matter, please do not hesitate to contact us.

Sincerely,

W Gary Gates

for K. J. Morris
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KJM/rh

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CLARIFICATION OF INFORMATION RELATING TO THE OPERATION OF THE INTERNALS VIBRATION MONITORING SYSTEM

This report is in response to the Safety Evaluation performed by the office of Nuclear Reactor Regulation relating to the threshold levels developed for the Fort Calhoun Station Internals Vibration Monitoring (IVM) system. The purpose of this report is to provide additional information on internals support degradation events at other Combustion Engineering facilities, the methodology used for developing IVM threshold levels, and the analysis of this data. This additional information was requested in the July 14, 1988 NRC Safety Evaluation of the proposed threshold levels.

The body of the report addresses the questions raised in the NRC Safety Evaluation. This information was discussed in a telephone conversation held between OPPD and the NRC on June 27, 1988. Attachment 1 is included to provide additional information on the low power IVM data acquisition at Fort Calhoun. Attachment 2 is included to provide the basis of the Fort Calhoun IVM threshold levels.

OPPD believes that the submitted threshold levels are able to detect early signs of degradation of the thermal shield support mechanism. It is requested that the NRC perform a reevaluation of the proposed threshold levels using the following report for supporting information.

INTRODUCTION

The Omaha Public Power District (OPPD) submitted the Fort Calhoun Thermal Shield Support System Inspection Deferral report to the NRC on August 28, 1986 (Reference 1). The purpose of the deferral report was to replace the commitment for a 1987 thermal shield inspection with an internals inspection to be performed during the 1993 outage. The NRC safety evaluation of the deferral report, dated February 12, 1987, accepted the deferral request contingent upon OPPD supplying additional information on the Internals Vibration Monitoring (IVM) system and developing IVM threshold levels (Reference 2). On October 13, 1987, OPPD submitted the additional information (Reference 3) and the February 25, 1988 submittal (Reference 4) proposed the Fort Calhoun IVM threshold levels. The NRC Safety Evaluation of the threshold levels, dated July 14, 1988 (Reference 5) determined that the proposed threshold levels were not acceptable. It is the intent of this report to provide clarification and additional supporting information in order to support OPPD's position that the proposed threshold levels are valid and acceptable.

Two questions were posed in the Safety Evaluation of the IVM threshold levels, they were:

1. Is the method on which the Fort Calhoun IVM is based reasonable?

2. Are the proposed threshold levels for detecting thermal shield support degradation reasonable?

OPPD believes that both the methodology and the proposed threshold levels are valid. This report will explain the basis for OPPD's position on these questions.

ITEMS OF CLARIFICATION

Before directly addressing the two questions, OPPD would like to clarify some important items that were referenced to in the subject Safety Evaluation.

- ST. LUCIE VISUAL INSPECTION

  One of the concerns raised by the NRC in Section 2.1 and Section 2.3 of the threshold level evaluation relates to the statement that the St. Lucie experience indicates that visual inspections may not be adequate to determine the condition of the thermal shield supports. The NRC evaluation states, "Neither visual inspection detected degradation of the thermal shield supports." The inspections referred to at St. Lucie were performed in 1978 and in 1981 with the core support barrel and thermal shield in place inside the reactor vessel. The two examinations performed included a remote visual examination of the upper guide structure and only accessible parts of the core support barrel. The St. Lucie examinations were performed in an attempt to locate loose parts within the reactor vessel. As discussed in Reference 7, the thermal shield was not included in the scope of the inspection and thus degradation of the supports was not detected.

- FORT CALHOUN VISUAL INSPECTION

  Section 2.3 of the safety evaluation compares the St. Lucie visual examination to the Fort Calhoun 1983 inservice inspection of the reactor internals. As discussed above and presented in Reference 7, the St. Lucie visual inspection was not a 10 year inservice inspection, and the Thermal Shield was not listed in the scope of their inspection. The emphasis of the 10 year ISI at Fort Calhoun was placed on inspection of the thermal shield positioning pins, locking collars, and lock welds. The core support barrel and the thermal shield were placed in a lower portion of the refueling cavity for this inspection. As a result of this inspection of the core support barrel and thermal shield, OPPD and the inservice inspection contractor concluded that these components were in the as-built condition.

  Section 6.2.2 of the August 28, 1986 OPPD submittal (Reference 1) provides a detailed description of the thermal shield support inspection performed in 1983 and the results which support OPPD's conclusions.

- FORT CALHOUN CURRENT INTERNALS CONDITION

  OPPD believes that the Fort Calhoun Station reactor internals are currently in the as-built condition. This statement is based on the following three items:

  1) 1983 ISI examination of the reactor internals.

  2) Long term frequency behavior of the 12.5 Hz resonance.

  3) Low power data analysis.

o FORT CALHOUN CURRENT INTERNALS CONDITION (Continued)

The first supporting point in assessing the present condition of the reactor internals is the 1983 ISI examination of the reactor internals. This examination concluded that the core support barrel and the thermal shield were in the as-built condition. The previous section on the Fort Calhoun Visual Inspection should be referenced for additional information.

The second supporting point is the long term frequency behavior of the 12.5 Hz resonance. The 12.5 Hz resonance is the primary indicator of any change in the adequacy of the reactor internals support mechanism.

The long term frequency behavior of the 12.5 Hz resonance was discussed in detail in Section 6.3 of the initial deferral report (Reference 1). Neutron noise data was analyzed and presented from the period of 1974 through 1986. As stated in the Reference (1) report, Section 6.3.2 Conclusions, "Therefore, based on excore neutron signal evaluations, thermal shield support system conditions have not degraded at full power conditions."

The final item supporting the reactor internals current condition is the low power neutron noise data collected at Fort Calhoun. The low power data identifies any changes in the mechanical preload of the reactor supporting mechanism and provides the earliest warning of a possible change in the effectiveness of the positioning pins.

A number of factors influence the effectiveness of the positioning pins. The positive factors, those which increase the coupling between the core support barrel and the thermal shield, are the initial mechanical preload and the differential thermal expansion force which increases with reactor power levels. If the reactor approaches an isothermal condition (i.e., low power), the differential thermal expansion force approaches zero and the initial mechanical preload would be the sole force maintaining positive load on the positioning pins. By comparing the low power data to the full power excore data, an evaluation can be made as to the adequacy of the initial mechanical preload and any changes in the preload from previous cycles.

The use of this technique at Fort Calhoun is further described in Attachment 1, the SMORN V paper "The Use of Excore Neutron Noise at Near Zero Reactor Power to Monitor Thermal Shield Support System Integrity." OPPD periodically performs low power IVM measurements when returning to power operation following a refueling outage. As described in the SMORN paper, evaluation of low power data at Fort Calhoun has shown no indication of degradation of the thermal shield support system.

Based on the previously discussed items, OPPD believes that thermal shield support degradation has not taken place and that the thermal shield supports are currently in the as-built condition.

o <u>ORNL POST-FAILURE ANALYSIS</u>

The NRC Safety Evaluation of the proposed threshold levels used the results of the post-failure analysis performed by Oak Ridge National Laboratory (ORNL) of neutron noise data taken from St. Lucie. The ORNL post-failure report indicated that the degradation of the thermal shield supports was manifested by small shifts (less than 2 Hz) in the frequency of several resonances in the neutron noise and the largest frequency shifts occurred during the second cycle. It is believed that this led ORNL to conclude that thermal shield support degradation occurred early in plant life at St. Lucie.

The ORNL analysis, reported in ORNL/NRC/LTR-85/24, was based on changes viewed in IVM Power Spectral Density plots over time. The ORNL analysis identified a small frequency change (less than 2 Hz) in a frequency as being the identifier of support degradation. OPPD, along with Combustion Engineering, believes that the frequency selected by ORNL is not related to core barrel motion.

OPPD believes that the ORNL report was used as the basis for the NRC conclusions of the threshold levels of the Fort Calhoun IVM program and does not believe that this is the correct conclusion. The following is a discussion of what OPPD and CE believe is the correct post-failure analysis of the St. Lucie event.

Combustion Engineering (CE) report CEN-272(F)-P (Reference 6) utilized calculated natural frequencies and mode shapes of the coupled core support barrel and thermal shield system for St. Lucie to determine changes in the vibration characteristics with assumed changes in the condition of the thermal shield support system. A comprehensive finite element model was developed to analyze different support cases such as; as-built conditions, with positioning pins removed, and with certain support lugs removed. Significant changes in the cos 2θ shell mode of vibration were calculated with simulated support system degradation.

In the as-built condition, the supports fully couple the thermal shield to the core support barrel causing the system to achieve approximately the characteristics of the core support barrel alone. For the no positioning pins condition, the bottom of the thermal shield is uncoupled from the core support barrel, allowing the partially coupled thermal shield to tend towards its own characteristics. With lug damage the two components uncouple, allowing the thermal shield to achieve approximately the characteristics of the thermal shield alone. The frequencies for the cos 2θ shell mode for these three support cases are 7.6 Hz, 5.1 Hz, and 3.1 Hz, respectively. For the same three support condition cases, the cantilever beam mode of the core support barrel and the thermal shield remains essentially constant: 6.8 Hz, 6.7 Hz, and 6.7 Hz.

By comparing the cos 2θ shell mode frequencies calculated using the finite element model to the actual operational data taken from St. Lucie, the different types of support mechanism degradation states can be traced. The results indicate that the positioning pins had not lost their effectiveness until the beginning of Cycle 4 and that support lug wear was first evident in the middle of

- ORNL POST-FAILURE ANALYSIS (Continued)

Cycle 5. The cos 2θ shell mode frequency did not change in the second cycle as reported by ORNL. Also, the frequency shifts of the cos 2θ mode associated with degradation of the thermal shield supports was manifested by rather large shifts (between 2.5 and 4.5 Hz) in frequency.

For the above reasons, OPPD believes that the ORNL report conclusions should not be used in the evaluations of the proposed threshold levels. OPPD utilizes the same finite element analysis that was performed for St. Lucie in developing the IVM threshold levels for the Fort Calhoun Station which is discussed in detail in Attachment 2.

- IVM AND LPM EVALUATION

The NRC Safety Evaluation pointed out two items of concern about IVM and LPM evaluations performed at the Fort Calhoun Station. The first item deals with the RMS threshold values calculated in the IVM analysis while the second item deals with the frequency of loose parts monitoring data analysis.

One item under Section 2.3 of the Safety Evaluation discusses the derivation of RMS threshold values. This section of the evaluation implies that OPPD only performs RMS level determinations of the neutron noise data. OPPD performs a complete analysis of each set of neutron noise data that is acquired. This data analysis includes but is not limited to RMS threshold levels. Power Spectral Densities, Cross Power Spectral Densities, Coherence, and Phase plots are used as other methods of analysis along with the Phase Separated Power Spectral Densities from which the RMS levels are determined. Additional information on the data analysis and the RMS threshold value derivation can be found in Section II of the OPPD report cited in Reference 3.

A second related item that warrants clarification is the frequency of loose parts monitoring data analysis. The NRC evaluation stated, "In the proposed surveillance of neutron noise, LPM data is utilized only after six successive months of monitoring indicate a loss of effectiveness of positioning pins." This statement is not true and can be clarified by reviewing the previous OPPD submittals to the NRC. The original August 28, 1986 submittal (Reference 1) and the February 25, 1988 threshold level submittal (Reference 4) clearly states that LPM data is analyzed on the same quarterly schedule as IVM data. OPPD utilizes LPM data on the same schedule and in conjunction with IVM data to obtain the necessary information to access the current internals condition.

- ITEMS OF CLARIFICATION SUMMARY

The previous five items were clarified due to the items recurring in the NRC Safety Evaluation. The clarification will now allow OPPD to directly address the two questions raised by the NRC.

IS THE METHOD ON WHICH THE FORT CALHOUN IVM IS BASED REASONABLE?

The NRC Safety Evaluation questions five key areas in assessing the effectiveness of detecting internals degradation. Each of the five areas will be addressed in detail in the following discussion. OPPD believes that there is solid evidence that Fort Calhoun has the required expertise to complement the IVM hardware and software to effectively diagnose thermal shield support degradation. This statement is based on the following five areas.

- THE TRAINING AND EXPERIENCE OF THE OPERATOR

  The NRC safety evaluation questions the training and experience of the personnel responsible for the Fort Calhoun Station LPM and IVM programs. Combustion Engineering is responsible for completing the analyses with OPPD personnel evaluating the results.

  The individual from OPPD responsible for the LPM and IVM programs at Fort Calhoun has over four years of nuclear plant experience and has attended the previous three U.S. reactor diagnostic conferences. The theme of these conferences is new LPM and IVM technologies. The individual has completed participation in the Combustion Engineering Owners Group program for LPM and IVM education. The Owners Group developed good practice manuals for both programs and also provided in-depth training on both technologies. The individual coauthored the attached technical paper presented at the 1987 SMORN conference entitled, "The Use of Excore Neutron Noise at Near Zero Reactor Power to Monitor Thermal Shield Support System Integrity."

  As part of the recent reorganization of the nuclear operations departments, OPPD has assigned this individual as the System Engineer responsible for the IVM and LPM systems. This assignment is an assurance that OPPD is committed to providing the time and resources necessary to maintain and improve the LPM and IVM programs at Fort Calhoun.

- THE EXISTENCE OF A SUITABLE "BASELINE" LIBRARY OF SPECTRA

  The "baseline" spectra library for Fort Calhoun was presented in the August 28, 1986 OPPD submittal (Reference 1). Section 6.3 of this report examines neutron noise data acquired at Fort Calhoun during the period from 1974 through 1986. The SER states that Maine Yankee recorded data monthly for only one fuel cycle. OPPD believes that the "baseline" library of spectra for Fort Calhoun is more extensive and that the August 28, 1986 report should be referenced for additional information on the subject.

- ADEQUATE ADMINISTRATIVE CONTROLS

  The Fort Calhoun Station uses adequate administrative controls to ensure high quality data, periodic engineering staff review, and transmittal of results to plant management. The LPM and IVM procedures are administered by the Preventive Maintenance program at Fort Calhoun. The procedures are routinely performed by qualified technicians at the station to assure high quality of the data.

o ADEQUATE ADMINISTRATIVE CONTROLS (Continued)

The procedure also requires that the results of the data analysis be transmitted to the plant manageme .c for review.

o DIAGNOSTIC PROCEDURES

The diagnostic procedures used for investigating changes in the noise signals that are not described by the baseline library are stated in the threshold level submittal (Reference 4). The diagnostic procedures propose to increase monitoring of IVM data from a quarterly basis to a monthly basis. This increased surveillance allows for adequate trending of the threshold levels and provides a status of the degree of support mechanism degradation. The levels of support mechanism degradation are directly related to the RMS threshold levels to be discussed in detail in Attachment 2.

The diagnostic procedures also allow for increased monitoring of loose parts monitoring data. The LPM data is analyzed on the same schedule as IVM data.

o PLANT MANAGEMENT ACTION

The administrative controls require that the results of the IVM and LPM analysis be transmitted to the plant management. Any changes in the IVM or LPM data will be noted and appropriate actions recommended. The appropriate actions, such as increased monitoring for diagnostic purposes or plant shutdown for a thermal shield support inspection, will be recommended as stated in the threshold level submittal (Reference 4).

ARE THE PROPOSED THRESHOLD LEVELS FOR DETECTING THERMAL SHIELD SUPPORT DEGRADATION REASONABLE?

This section of the NRC safety evaluation raises several questions on the reasonableness of the proposed threshold levels for detecting possible thermal shield support degradation. Several of these questions, such as the long-term frequency behavior of the 12.5 Hz resonance and the lack of coordination between the LPM and IVM systems, have already been answered in the Items of Clarification section of this report. Other questions raised in the safety evaluation, such as the questioning of the 100% increase in RMS levels when degradation occurs and the reasoning behind basing the threshold levels on the 6 to 10 Hz range, will be answered in Attachment 2. Attachment is a detailed description of the methodology used to develop the Fort Calhoun threshold levels.

The purpose of this section is to provide a brief summary of the method used to develop the Fort Calhoun Station threshold levels.

The threshold level method used at Fort Calhoun is based on the experience gained during the failure mechanism analysis of the St. Lucie thermal shield. The basic approach used for the threshold levels are to monitor the cos 2θ shell mode of the core support barrel and thermal shield. For the as-built condition, the frequency of this mode is calculated and shown in actual operational data to be at 12.5 Hz. For the condition of loss of effectiveness of the positioning pins, the frequency of this mode is calculated to be 7.9 Hz.

When the 12.5 Hz mode disappears, the power at that frequency mode will appear at 7.9 Hz. The 7.9 Hz in-phase mode would tend to be hidden by the out-of-phase beam mode at 7 Hz in a standard Power Spectral Density (PSD) plot. By eliminating the out-of-phase portion of the PSD using established phase separating techniques, the cos 2θ in-phase mode can be easily tracked.

Looking for the disappearance of the cos 2θ shell mode at one frequency and its reappearance at a different frequency provides the most meaningful method for monitoring the condition of the thermal shield support system. To provide a threshold value for monitoring in response to the NRC request, the power associated with the as-built cos 2θ in-phase mode was quantified using previous IVM operational data. The frequency associated with degraded thermal shield support conditions for the cos 2θ mode were calculated using a detailed finite element model. The power associated with degraded thermal shield support conditions is assumed to remain constant and shift to the calculated frequencies. Based on the comparison of the two cases, a percent of change in RMS levels for the 6 to 10 Hz range was devised. The acceptable and unacceptable percent of change levels are explicitly stated in the threshold level submittal (Reference 4).

Attachment 2, Fort Calhoun IVM Threshold Level Development, should be referenced for additional information on the development of the threshold levels for the Fort Calhoun Station.

CONCLUSION

Based on the cases presented in this report, the Fort Calhoun Station noise monitoring program would be able to detect degradation of the thermal shield supports before any damage took place. OPPD possesses the expertise and administrative controls necessary to ensure an effective IVM program. OPPD has addressed the specific points raised in the NRC safety evaluation and requests the NRC acceptance of the Fort Calhoun Station IVM threshold levels.

REFERENCES

1. Letter from R. L. Andrews (OPPD) to USNRC, "Fort Calhoun Thermal Shield Support System Inspection Deferral," dated August 28, 1986 (LIC-86-421).

2. Letter from W. Paulsen (USNRC) to R. L. Andrews (OPPD), "Thermal Shield Support System Inspection Deferral, Fort Calhoun Station, Unit No. 1," dated February 12, 1987.

3. Letter from R. L. Andrews (OPPD) to USNRC, "Additional Information on the Fort Calhoun Internals Vibration Monitoring System," dated October 13, 1987 (LIC-87-673).

4. Letter from R. L. Andrews (OPPD) to USNRC, "Threshold Levels for the Fort Calhoun Internals Vibration Monitoring System," dated February 25, 1988 (LIC-88-091).

5. Letter from P. D. Milano (USNRC) to K. J. Morris (OPPD), "Request for Additional Information Relating to the Operation of the Internal Vessel Monitoring (IVM) System," dated July 14, 1988.

6. CEN-272(F), "Final Report on the St. Lucie Unit 1 Post Cycle 5 Plant Recovery Program," dated February 1984.

7. Combustion Engineering Report TR-FIS-017, "Florida Power and Light, St. Lucie Unit, Reactor Internals Inspection Report, dated March 25, 1982.

ATTACHMENT 1

The Use of Excore Neutron Noise at Near Zero Reactor Power to Monitor Thermal Shield Support System Integrity

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Joel A. Stevens — Combustion Engineering, Inc.

Keywords: Thermal shield, excore detectors, neutron noise, near isothermal conditions, phase separated PSD.

Abstract

Several nuclear reactors which incorporate a thermal shield design have experienced degradation of the thermal shield support structure. Combustion Engineering, Inc. (C-E) and the Omaha Public Power District (OPPD) have developed both equipment and a surveillance program which monitors the performance of the thermal shield support structure at the OPPD Fort Calhoun nuclear station.

Background:

Both the ASME and the American National Standards Institute (ANSI) in their joint standard have accepted a consistent program of Internals Vibration Monitoring (IVM) using neutron noise to be a reliable reactor internals surveillance method (Ref. 1). The U. S. utility, OPPD, has developed an administrative procedure to monitor reactor internals vibration levels using existing excore neutron detector signals at the Fort Calhoun station, located north of Omaha, Nebraska.

The Fort Calhoun reactor (Figure 1) is a 502 MWe Pressurized Water Reactor (PWR) designed by C-E. In operation since late 1973 the reactor internals include a Core Support Barrel (CSB) and a Thermal Shield (TS). During the scheduled ten year In-service Inspection (ISI) performed in 1983 a visual examination was made of the CSB/TS support structures. The examination indicated that all support components were in good condition. OPPD has been actively involved in a regular program of IVM surveillance of reactor internals motion since 1985.

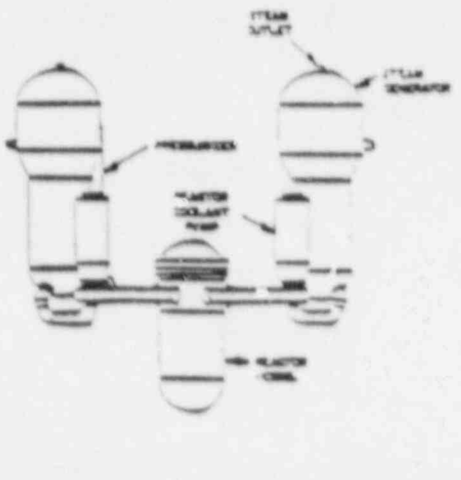


Figure 1. Ft. Calhoun Reactor

Core Barrel Thermal Shield Description:

Figure 2 details the arrangement of the Ft. Calhoun reactor internals. The thermal shield is a right circular cylinder concentric with the core barrel. The thermal shield is attached to the core barrel and extends the length of the active core. The thermal shield is suspended from support lugs on the core support barrel. Radial positioning is provided by two sets of positioning pins. Figure 3 describes the CSB/TS connection scheme.

CSB/TS Dynamic Behavior:

Because of the support lugs and the radial positioning pin the CSB/TS form a closely coupled mechanical structure which exhibits identifiable beam

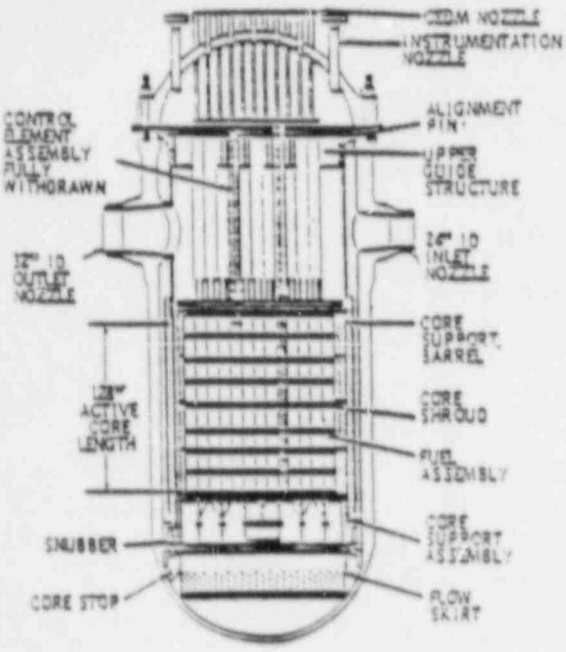


Figure 2. Ft. Calhoun Reactor Internals

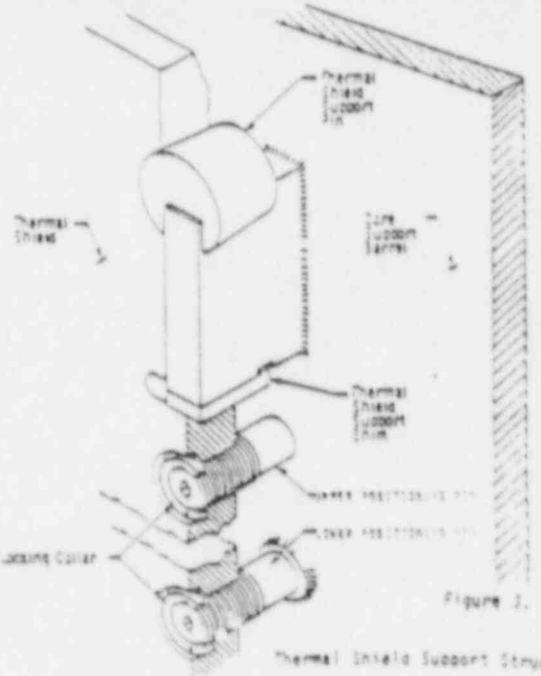


Figure 3.

Thermal Shield Support Structure

and shell modes of vibration. The beam bending mode (Figure 4a) is a cantilever mode of vibration of the CSB, similar to a simple beam with one end free and one end clamped. In this mode the CSB cross section remains circular and translates. The shell modes of vibration (Figure 4b, 4c) are vibration modes involving circumferential variation in the shape of the CSB.

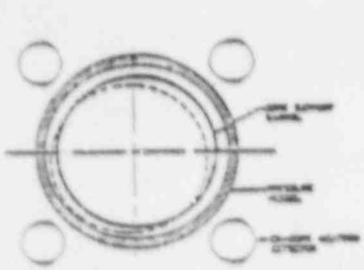


Figure 4A.

Beam Mode Vibration

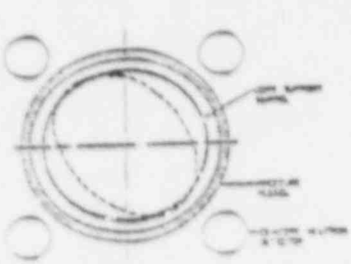


Figure 4B.

First Shell Mode (COS 2θ).

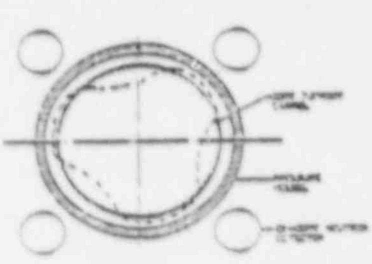


Figure 4C.

COS 2θ Shell Mode Vibration

TABLE 1

Predictions of In-Water Modal Frequencies (Hertz)

	Mode	Nominal	All Pins Removed
1.	Beam	7	7
2.	cos 2θ	12.5	7.9
3.	cos 3θ	16.3	14.9
4.	cos 4θ	22.8	22

Column #1 in Table 1 lists the calculated in-water modal frequencies of the Fort Calhoun CSB/TS (Ref. 2). These frequencies can be identified in a typical Auto Power Spectral Density (APSD) taken from the Ft. Calhoun reactor at 100% power (Fig. 5).

Analysis of the CSB/TS support structure (Ref. 3) indicates that a loss of radial positioning pin effectiveness can initiate the decoupling of the CSB/TS. This decoupling effects vibration modal frequencies. The detection of radial positioning pin condition is possible using the neutron noise portion of the existing linear power range detectors.

Column #2 of Table 1 lists the calculated in-water modal response frequencies for the Ft. Calhoun CSB/TS in the case of a postulated loss of effectiveness of all radial positioning pins. Based on comparison between "Nominal" and "All pins removed" cases, it can be observed that only the first shell mode of vibration (COS 2θ) changes significantly from its nominal 12.5 Hz down to 7.9 Hz with all pins removed. By monitoring the frequency of the CSB/TS first shell mode one can infer changes in positioning pin effectiveness.

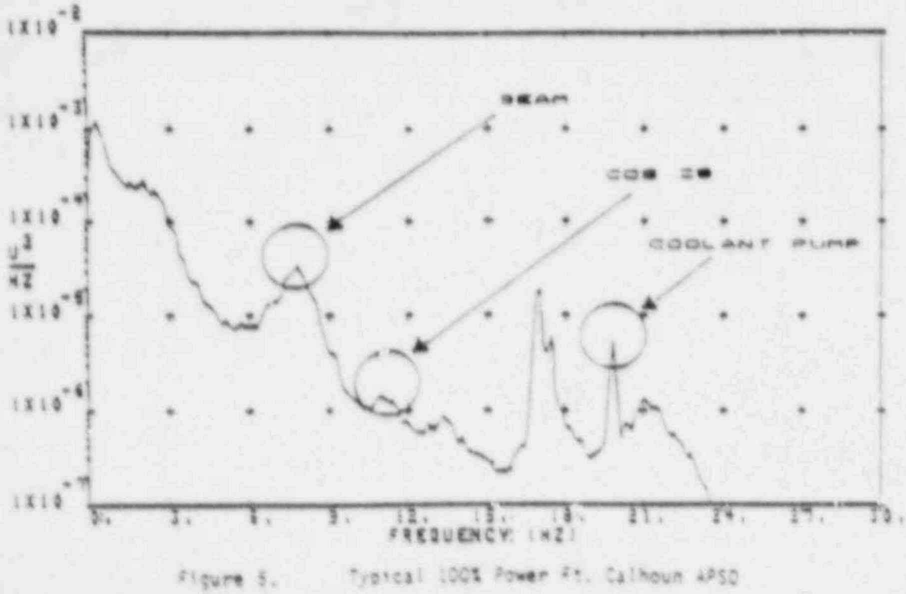


Figure 5. Typical 100% Power Ft. Calhoun APSD

Reactor Power and Positioning Pin Loading:

A number of factors influence the effectiveness of the radial positioning pin including mechanical preload and differential thermal expansion, both of which increase the coupling force between the thermal shield, and the CSB. The pressure differential across the thermal shield, support lug bending and radiation induced relaxation all decrease the CSB/TS coupling force. The thermal expansion force is the only positioning pin loading force which can be controlled during reactor operation. Increased temperature differential between the core support barrel and the thermal shield increases the interference between the positioning pin and the CSB.

Power Level Selection:

In order to monitor positioning pin effectiveness it was desired to select plant operating conditions for which the CSB/TS would be most susceptible to any reduction in positioning pin effectiveness. If the reactor is in an isothermal condition the thermal expansion force is zero and the mechanical preload given the positioning pins during installation would be the sole force maintaining a positive load on the positioning pins.

Zero rector power produces ideal isothermal conditions between the CSB/TS components. However, zero reactor power does not provide an adequate neutron flux to allow the linear power range excore detectors to be used. If a simple measurement scheme using existing instrumentation is desired then a near zero power level must be selected which:

1) Produces a minimal thermal expansion force.
2) Provides an adequate excore detector flux signal.

The thermal expansion force exerted by the CSB on the positioning pins is:

$$F_{therm} = K \times (R_o + {}^{L}/2) \times \alpha \times \Delta T$$

K = CSB/TS stiffness
R_o = Core barrel outer radius
L = Positioning pin length

α = Coefficient of thermal expansion for the core barrel- thermal shield.
ΔT = Temperature difference between CBS and TS.

Since ΔT is nearly linear with reactor power the thermal expansion force is also nearly linear with reactor power. Using this linear relation the thermal expansion force at 5% reactor power would only be 1/20th the 100% thermal expansion force. From a thermal expansion force consideration the lower the reactor power the better. Reactor power below 5% does not produce significant thermal expansion force.

The other consideration in selecting a reactor power level was would there be sufficient vibration related neutron noise signal to allow a successful measurement using the plant linear power range detector signals. The background noise level is made up of uncorrelated noise sources, not related to the vibration induced fluctuations, found anywhere along the excore detector signal path. This background level has an APSD which is constant at all frequencies and has a constant voltage value. The vibration related fluctuations measured by the excore detectors are linear with reactor power. Our measured signal is their sum of:

X (signal) = vibration (power) + BACKGROUND (constant)

Since the background level is constant it becomes a progressively larger percentage of the measured signal as the reactor power level decreases. Figure 6 is a composite of APSD's for three different power levels (30%, 10% and 5%). In all three traces the vibration portion of the signal remains a relatively constant percentage of the D.C. power. The background noise is constant regardless of reactor power level and therefore consistently becomes a larger percentage of the signal as the power level decreases.

A problem exists with measuring shell mode frequencies. The shell mode vibration scale factor, (% neutron flux/mil of motion), is smaller by a factor of three (3) than the scale factor for the beam mode (Ref. 4). Coupled with this smaller shell mode scale factor is the smaller shell mode vibration amplitudes. Together these two factors produce a detected shell mode neutron flux signal which is about a factor of one hundred (100) smaller than the beam mode response in the APSD. Examining Figure 6 at the beam and shell frequencies it is quite easy to identify the beam mode frequency at all three power levels. However, it becomes increasingly difficult to identify the first shell mode as the reactor power decreases.

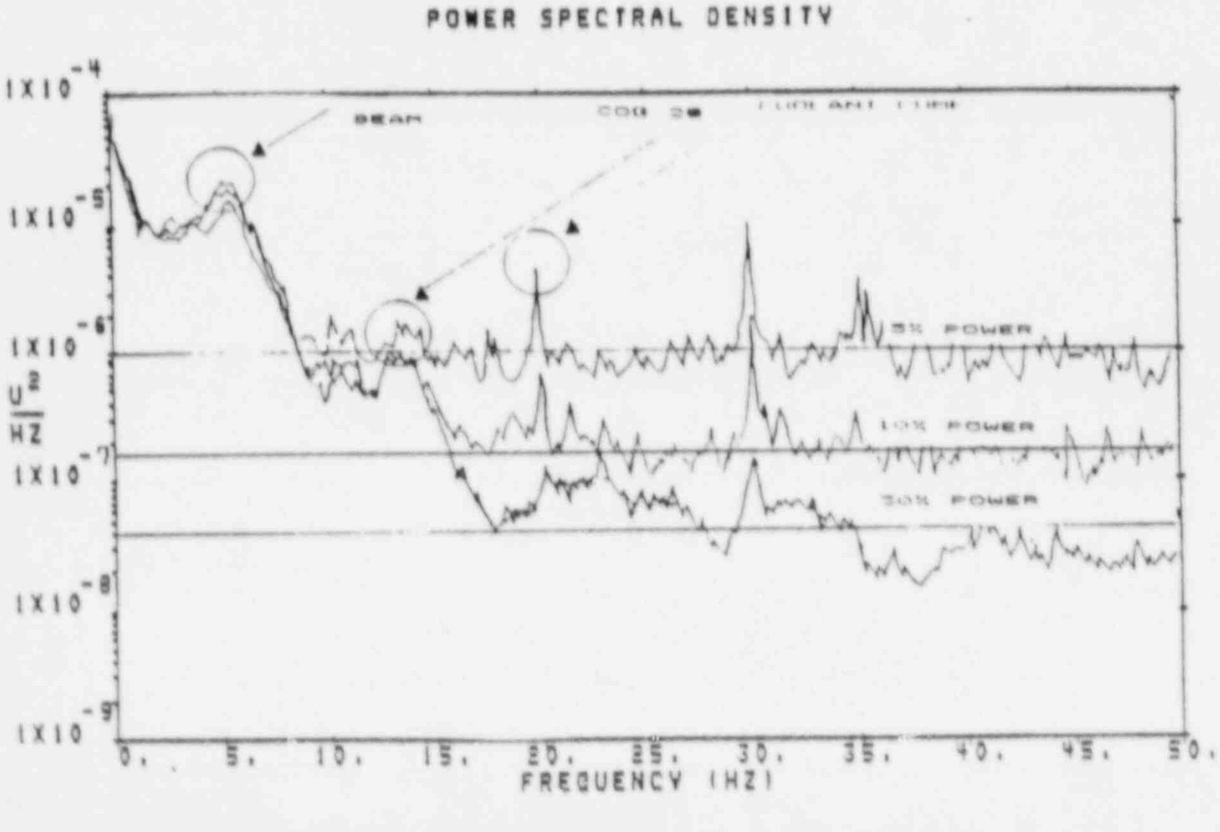


Figure 6 Composite APSD Spectra - versus - Power Level

Five percent reactor power was chosen to be the plant power level for the positioning pin effectiveness measurement. The five percent power level is both the highest power level at which thermal conditions are not appreciable and the lowest reactor power at which CSB/TS shell mode frequencies can be detected.

Data Acquisition System:

The measurement of reactor internals vibration data requires specialized signal conditioning equipment. C-E has developed an Internals Vibration Monitoring (IVM) system which provides all of the necessary electronic equipment and analysis functions needed to perform regular IVM measurements. The IVM system contains a Signal Conditioning Module (SCM) which houses the processing electronics, a portable computer, which controls the SCM and computes the analysis functions, and a graphics hard copy printer.

The Signal Conditioning Module accepts two channels of excore signals as inputs and outputs the bandlimited and amplified vibration fluctuations which make up less than 0.25% of the gross power level excore signal.

The C-E SCM (Fig. 7) provides:

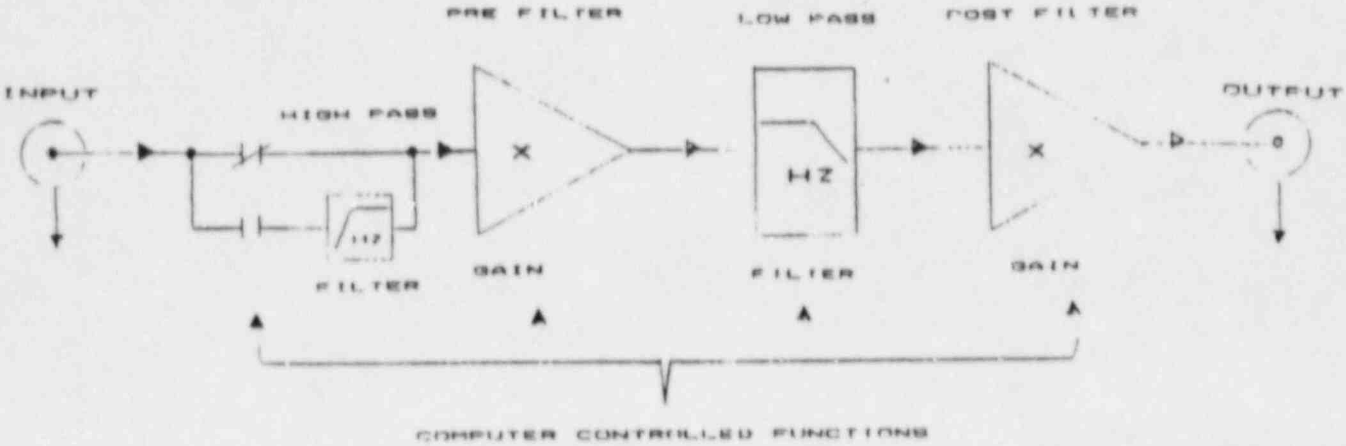
- o High pass filtering for elimination of the gross DC power level.
- o High gain PRE-filtering amplification
- o Low pass filtering for frequency limiting the data
- o High gain POST-filter amplification
- o Controlled sample rate Analog - to - Digital Conversion (ADC).

The SCM is totally controlled in all its ranges and functions by the computer.

The output of the SCM is then processed by the IVM analysis software package and analysis results are stored on both disk and hardcopy graphic printer output.

The C-E IVM analysis software acquires both time and frequency domain data. The standard IVM analysis functions include: Auto and Cross Power Spectral Density functions, signal Coherence and Relative Phase functions as well as user specified band select RMS (root-mean-square) energy calculations. Also available is the implementation of spectral phase separation algorithms. The IVM software can also recall previous computed analysis results to allow comparison between two analysis data sets.

Figure 7. C-E Internals Vibration Monitoring System Signal Conditioning Module Block Diagram



Near Zero Power Field Measurement

In June 1986 a near zero neutron noise measurement was made at Ft. Calhoun plant. This measurement was made at nominally 5% reactor power using the C-E IVM system. Two linear excore detectors 180° apart were measured and analyzed. The signal obtained from the first shell mode exhibits an in-phase relationship between detectors which are 180° apart (Fig. 8). In addition to the normal spectral analysis a special phase separated APSD (Ref. 5) was calculated. This phase separation technique applies to the APSD processing for which the measured APSD is the sum of two processes which are either in phase (0°) or out-of-phase (180°) relative to each other.

For this special case the measured APSD can be separated based on phase into APSD (0°) and APSD (180°). Since our interest was in the CSB/TS first shell mode frequency, the in-phase, APSD(0°) spectra, would be the most sensitive to change of that vibration mode. Figure 9 is the 0°-phase APSD from the 5% reactor power measurement at Ft. Calhoun. The frequency of this vibrational mode is still noted to be in the same 10-12 Hertz region as in the 100% power APSD (Fig. 5). The inference from this analysis was that the CSB/TS support system was effective.

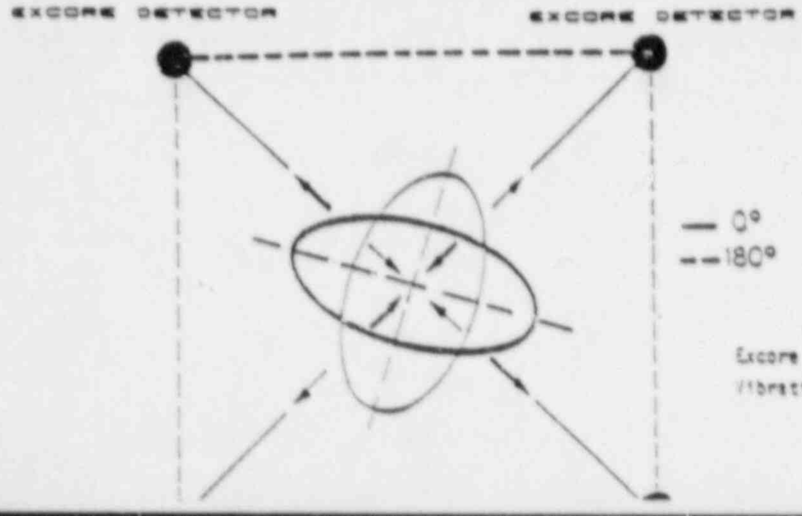


Figure 8.

Excore Detector Phase Relationships for First Shell Mode [illegible] Vibration

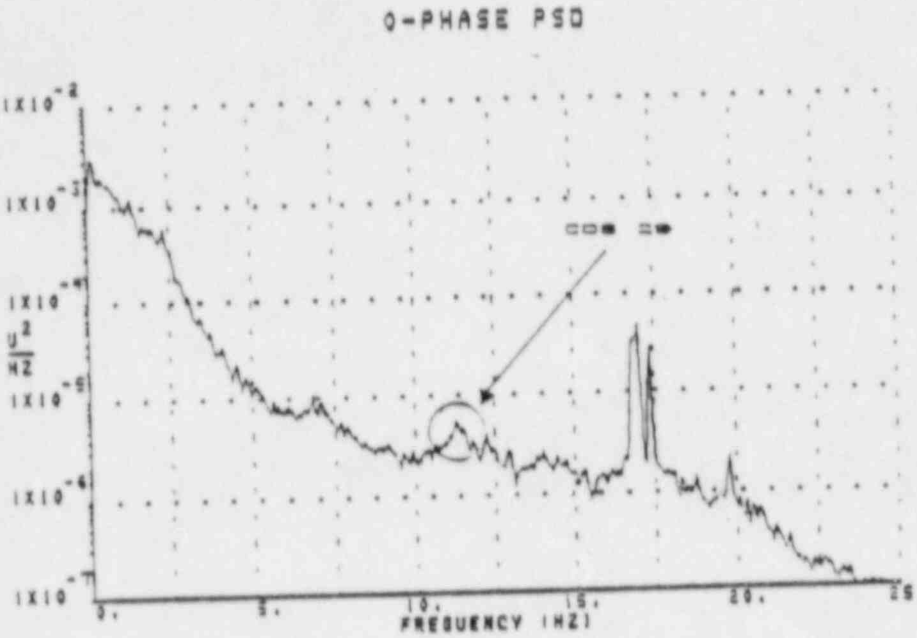


Figure 9. Ft. Calhoun 0° Phase APSD at 5% Reactor Power.

Inspection Deferred Through Monitoring

In 1984, the U. S. Nuclear Regulatory Commission (USNRC) concern over CSB/TS support system problems led OPPD to commit to the USNRC to perform an examination of the Ft. Calhoun CSB/TS during the 1987 maintenance outage. This examination would have been performed four years after the normal ISI ten year inspection. OPPD petitioned the USNRC in August 1986 for deferral of this examination until the next scheduled ISI inspection to be performed in 1993.

The deferral effort consisted of both an analytical effort and a commitment to an IVM surveillance program. Emphasis in the deferral effort was also given to the positive results of the near zero power IVM measurement taken in June 1986.

In February 1987 the USNRC granted OPPD a deferral from CSB/TS examination until the scheduled 1993 ISI date. OPPD's commitment to the USNRC included the analysis of IVM data on a quarterly basis and collecting near isothermal IVM data once per fuel cycle.

Conclusion

The early detection of thermal shield support system structural change is the primary goal of the OPPD IVM program at Ft. Calhoun. IVM measurements at near zero reactor power provide an early warning of a possible change in radial positioning pin effectiveness by monitoring changes in the CSB/TS vibration characteristics. The OPPD surveillance program provides a reliable early warning of potential structural problems with a minimal impact on plant operations.

References

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3. Lubin, B., Longo et al, "Analysis of Vibration Monitoring and Loose Parts Data Related to the St. Lucie 1 Thermal Shield Failure", Presented SMORN V 1987.

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ATTACHMENT 2

FORT CALHOUN IVM THRESHOLD LEVEL DEVELOPMENT

BACKGROUND

The technical approach utilized to develop the Fort Calhoun IVM threshold levels is based on the experience gained during the failure mechanism analysis of the St. Lucie 1 thermal shield (Reference 1). In the St. Lucie analysis program, vibration structural response calculations were performed on a detailed finite element model of the core support barrel, thermal shield and thermal shield supports as a coupled system. These calculations were performed for the system in its nominal as-designed condition and in assumed degraded conditions. A significant change in frequency for a particular shell mode of vibration was noted from the calculations with the simulated loss of all positioning pin effectiveness and further change occurred when simulated support lug damage was introduced into the calculation.

St. Lucie Loose Parts Monitoring (LPM) and Internal Vibration Monitoring (IVM) data was reviewed and reanalyzed to determine if there were quantifiable changes in the data during plant operation prior to actual degradation occurring. LPM data shows changes in both magnitude and frequency of the impact signals and changes in location of the signals with operating time. Phase separation analysis of the IVM data showed changes in an in-phase shell mode frequency with time. The initial frequency and the corresponding frequency reductions found in the degraded condition in the IVM data showed excellent correlation with the values calculated for the nominal and assumed degraded support system conditions.

The available evidence (summarized graphically in Figure 1) indicates that the thermal shield support system degradation process is lengthy, and that the process is detectable by the methods of excore neutron noise monitoring and loose parts monitoring systems.

FORT CALHOUN FREQUENCY CALCULATIONS

Experience gained from the St. Lucie Unit 1 program demonstrated that degradation in the thermal shield support system were manifested as frequency peak changes in the spectra of the excore detector noise signals. Therefore, analytic predictions of changes in frequencies and modes with assumed changes in the thermal shield support system are used to interpret the data acquired from the excore detector signal monitoring program.

A detailed three-dimensional finite element model of the core support barrel, thermal shield and thermal shield support system has been developed for Fort Calhoun (see Figure 2). Natural frequencies of the coupled system were calculated by means of the SAP 4 computer program for the nominal support system case, for the loss of effectiveness of all positioning pins case, and the removal of the structural restraint provided by four support lugs. The results of these frequency calculations are summarized in Table 1.

For Fort Calhoun, as was also the case for St. Lucie Unit 1, there are significant changes in frequency of the cos 2θ mode of vibration of the thermal shield and core support barrel system when simulated damage of the thermal shield support system is introduced into the analysis. This cos 2θ mode, which results in in-phase relationships between cross-core pairs of detectors, provides the requisite identifier of the condition of the thermal shield support system which can then be tracked via the IVM system.

FORT CALHOUN CYCLE 10 IVM DATA

Four sets of 100% power neutron noise data was analyzed for Fort Calhoun from Fuel Cycle 10. Data from the cross-core detector pairs B safety - C safety and B control - A control were acquired on 4/30/86, 5/28/86, 7/28/86 and 10/18/86. The four data sets were analyzed using phase separating techniques (Reference 2) over the frequency bands 0-4 Hz, 4-6 Hz, 6-10 Hz and 10-15 Hz. These bands were chosen so as to contain the Power Spectral Density (PSD) peaks corresponding to the initial (nominal condition) frequency (12.5 Hz) and the reduced (degraded condition) frequencies (7.9 Hz and 5.4 Hz) associated with the cos 2θ mode used to monitor the condition of the thermal shield supports.

The Root-Mean-Square (RMS) values of amplitude for the phase separated PSD's for the chosen frequency bands were calculated (see Table 2). The values listed in the columns headed 10-15 Hz provide an estimate, as a function of time for the various detectors, of the amplitude of the 12.5 Hz peak of the in-phase (0°) cos 2θ mode associated with the nominal condition of the thermal shield support system.

It is well known (e.g., see References 3, 4, 5 and 6) that neutron noise not related to internals vibration can vary throughout a fuel cycle and from cycle to cycle. These variations may be related to fuel burnup, soluble boron concentration, temperature and modifications in fuel management of design. The variations in the Fort Calhoun data during Cycle 10, calculated as a percent increase over the 4/30/86 data, are shown in Table 3. Note that the largest increase over any three month period for the 6-10 Hz band was 14.5% and that the largest increase over any one month period for the 4-6 Hz band was 6.3% ((27.8-8.9) ÷ 3).

ESTIMATE OF EXPECTED CHANGES IN IVM DATA WITH POSTULATED CHANGES IN THERMAL SHIELD SUPPORT SYSTEM CONDITIONS

Natural frequency calculations performed for Fort Calhoun, summarized in Table 1, show significant changes in the frequency of the cos 2θ mode of vibration of the thermal shield and core support barrel with postulated damage to the thermal shield support system. For the nominal, as-designed condition of the thermal shield support system, the frequency is 12.5 Hz; for the case of loss of effectiveness of all positioning pins, the frequency is 7.9 Hz; and for the case of support lug/support pin wear, the frequency is 5.4 Hz.

With the postulated loss of effectiveness of all positioning pins, the power associated with the 12.5 Hz peak will move to a peak at 7.9 Hz. With the postulated addition of support lug/support pin wear, the power originally associated with the 12.5 peak will decrease further to a peak at 5.4 Hz.

On this basis, estimates of expected changes in neutron noise data with the postulated changes in thermal shield support condition have been calculated. The results are given in Table 4, the case of loss of effectiveness of positioning pins (6-10 Hz band) and the case of support lug/support pin wear (4-6 Hz band). For example, for the C safety detector, the RMS amplitude of the 6-10 Hz band is estimated to increase from 100% to 117%, as a function of time, over its nominal value with postulated loss of effectiveness of all positioning pins. Also, for the same detector, the RMS amplitude of the 4-6 Hz band is estimated to increase from 43% to 51% over its nominal value with

FORT CALHOUN IVM THRESHOLD VALUES

A threshold value is proposed based on the following suggested approach: i.e., that a threshold value be chosen that reduces the likelihood of wear on the thermal shield support lugs/support pins. Experience at St. Lucie 1 indicated that approximately two years had passed between the time that loss of effectiveness of the positioning pins was detectable by IVM and the time that support lug/support pin wear had occurred. It is proposed, therefore, that a threshold value be selected that results in an inspection program about six months after detection of loss of effectiveness of all positioning pins. This can be done as follows:

1. Acquire, reduce, and evaluate 100% power IVM data at the beginning, end, and at three month intervals for each fuel cycle. Calculate the in-phase (0°) RMS amplitudes of the 0-4 Hz, 4-6 Hz, 6-10 Hz, and 10-15 Hz frequency ranges.

2. amplitude for the 6-10 Hz range for the present data to that of the previous quarterly data.

3. If there is a definite 12.5 Hz in-phase peak in the PSDs, and the increase in the RMS amplitude for the 6-10 Hz range is less than 25%, continue to monitor at three month intervals. If both of these conditions are met, the thermal shield is adequately supported. (The 25% increase is based on an expected three month increase associated with fuel burnup of 15% plus a potential 10% IVM system amplitude uncertainty).

4. If there is a greater than 25% but less than 100% increase in the RMS amplitude for the 6-10 Hz in-phase range from the previous quarterly data, begin acquiring, reducing, and evaluating data at one month intervals. The only action to be taken at this time is increased monitoring for trending purposes. No inspection is recommended at this time.

5. If there is no definite 12.5 Hz in-phase (0°) peak in the PSDs, and if the increase in the RMS amplitude of the 6-10 Hz range is greater than 100%, the changes in the neutron noise data indicate a loss of effectiveness of the positioning pins. Begin to monitor IVM data at one month intervals for trending purposes. An inspection of the thermal shield support structure is recommended. The inspection should occur within the next six months.

6. If six successive months of monitoring indicate a loss of effectiveness of positioning pins and if this is corroborated by evaluation of Loose Parts Monitoring (LPM) data, it will be recommended that the plant be shut down and an inspection of the thermal shield support structure be performed as soon as practicable.

LPM data is acquired, reduced and evaluated on the same schedule as that for IVM. Near-zero power IVM data is acquired, reduced and evaluated once per fuel cycle.

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2. "Detailed Neutron Noise Analysis of Pressurized Water Reactor Internals Vibration," Charles W. Mayo, Atomkernenergie 29 pp. 9-13 (1977).

3. "Analysis of Changes with Operating Time in the Calvert Cliffs Unit 1 Neutron Noise Signals," J. P. Steelman and B. T. Lubin, SMORN II, September 1977.

4. "Review of Borselle PWR Noise Experiments, Analysis and Instrumentation," E. Turkan, SMORN II, October 1981.

5. "Use of Neotron Noise for Diagnosis of In-Vessel Abnomalies in Light-Water Reactors," D. N. Fry, et. al., NUREG/CR-33303 (ORNL/TM-8774) January 1984.

6. "Contribution of Fuel Vibrations to Ex-Core Neutron Noise During the First and Second Cycles of the Sequoyah-1 Pressurized Water Reactor," F. J. Sweeney, et. al., SMORN IV, October 1984.

TABLE 1

FORT CALHOUN THERMAL SHIELD, CORE SUPPORT BARREL AND THERMAL SHIELD SUPPORT SYSTEM

IN-WATER MODAL FREQUENCIES (HERTZ) vs. SUPPORT SYSTEM CONDITION

Mode	Nominal	All Pins Removed	All Pins Removed & 4 Lugs Removed
BEAM	7	7	7
COS 2θ	12.5	7.9	5.4
COS 3θ	16.3	14.9	14
COS 4θ	22.8	22	21.3

TABLE 2

RMS AMPLITUDE OF PHASE SEPARATED PSD's FOR CHOSEN FREQUENCY BANDS

B SAFETY				
DATE (1985)	0° PHASE PSD (RMS$\times 10^3$) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	25.708	1.982	0.883	0.924
5-28	27.407	2.011	0.919	1.008
7-28	30.594	2.234	0.982	1.138
10-28	33.681	2.311	1.058	1.360

C SAFETY				
DATE (1986)	0° PHASE PSD (RMS$\times 10^3$) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	27.330	2.090	0.898	0.901
5-28	28.858	2.184	0.949	0.965
7-28	31.179	2.460	1.020	1.106
10-28	33.605	2.529	1.112	1.302

B CONTROL				
DATE (1986)	0° PHASE PSD (RMS$\times 10^3$) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	27.530	2.013	0.907	1.411
5-28	30.806	2.096	0.981	1.546
7-28	33.617	2.193	1.002	1.545
10-28	37.048	2.573	1.018	1.765

A CONTROL				
DATE (1986)	0° PHASE PSD (RMS$\times 10^3$) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	26.975	2.055	1.151	1.299
5-28	28.888	2.140	1.218	1.412
7-28	31.003	2.372	1.318	1.532
10-28	36.275	2.565	1.414	1.744

TABLE 3

PERCENT INCREASE IN RMS AMPLITUDE OF PHASE SEPARATED PSD's
FOR CHOSEN FREQUENCY BANDS

B SAFETY % CHANGE W/TIME				
DATE (1986)	0° PHASE PSD RMS INCREASE (%) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	-	-	-	-
5-28	6.6	0.9	4.1	9.1
7.28	19.0	12.7	11.2	23.2
10-28	31.0	16.6	19.8	47.2

C SAFETY % CHANGE W/TIME				
DATE (1986)	0° PHASE PSD RMS INCREASE (%) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	-	-	-	-
5-28	5.6	4.5	5.7	7.1
7-28	14.1	17.7	13.6	22.8
10-28	23.0	21.0	23.8	44.5

B CONTROL % CHANGE W/TIME				
DATE (1986)	0° PHASE PSD RMS INCREASE (%) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	-	-	-	
5-28	11.9	4.1	8.2	9.6
7-28	22.1	8.9	10.5	9.5
10-28	34.6	27.8	12.2	25.1

A CONTROL % CHANGE W/TIME				
DATE (1986)	0° PHASE PSD RMS INCREASE (%) FOR THE FREQUENCY BANDS			
	0-4Hz	4-6Hz	6-10Hz	10-15Hz
4-30	-	-	-	-
5-28	7.1	4.1	5.8	8.7
7-28	14.9	15.4	14.5	17.9
10-28	34.5	24.8	22.8	34.3

TABLE 4

PERCENT INCREASE IN RMS AMPLITUDE OF PHASE SEPARATED PSD's FOR CHOSEN FREQUENCY BANDS

B SAFETY 0° PHASE SEPARATED PSD						
DATE (1986)	"LOSS OF EFFECTIVENESS OF POSITIONING PINS" (6-10Hz)			"SUPPORT LUG/SUPPORT PIN WEAR" (4-6Hz)		
	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION
	NOMINAL CONDITION	DEGRADED CONDITION		NOMINAL CONDITION	DEGRADED CONDITION	
4-30	0.88	1.81	106	1.98	2.91	47
5-28	0.92	1.93	110	2.01	3.02	50
7-28	0.98	2.12	116	2.23	3.37	51
10-28	1.06	2.42	128	2.31	3.67	59

C SAFETY 0° PHASE SEPARATED PSD						
DATE (1986)	"LOSS OF EFFECTIVENESS OF POSITIONING PINS" (6-10Hz)			"SUPPORT LUG/SUPPORT PIN WEAR" (4-6Hz)		
	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION
	NOMINAL CONDITION	DEGRADED CONDITION		NOMINAL CONDITION	DEGRADED CONDITION	
4-30	0.90	1.80	100	2.09	2.99	43
5-28	0.95	1.91	101	2.18	3.15	44
7.28	1.02	2.13	109	2.46	3.57	45
10-28	1.11	2.4[illegible]	117	2.53	3.83	51

B CONTROL 0° PHASE SEPARATED PSD						
DATE (1986)	"LOSS OF EFFECTIVENESS OF POSITIONING PINS" (6-10Hz)			"SUPPORT LUG/SUPPORT PIN WEAR" (4-6Hz)		
	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION
	NOMINAL CONDITION	DEGRADED CONDITION		NOMINAL CONDITION	DEGRADED CONDITION	
4-30	0.91	2.32	155	2.01	3.42	70
5-28	0.98	2.51	158	2.10	3.64	73
7-28	1.00	2.55	155	2.19	3.74	71
10-28	1.07	2.79	173	2.57	4.34	69

A CONTROL 0° PHASE SEPARATED PSD						
DATE (1986)	"LOSS OF EFFECTIVENESS OF POSITIONING PINS" (6-10Hz)			"SUPPORT LUG/SUPPORT PIN WEAR" (4-6Hz)		
	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION	$RMS \times 10^3$		% CHANGE FROM NOMINAL TO DEGRADED CONDITION
	NOMINAL CONDITION	DEGRADED CONDITION		NOMINAL CONDITION	DEGRADED CONDITION	
4-30	1.15	2.45	113	2.06	3.35	63
5-28	1.27	2.63	116	2.14	3.55	66
7-28	1.37	2.86	117	2.37	3.90	65
10-28	1.41	3.16	124	2.57	4.31	68

CHRONOLOGICAL SEQUENCE OF EVENTS - ST. LUCIE 1

FIGURE 1

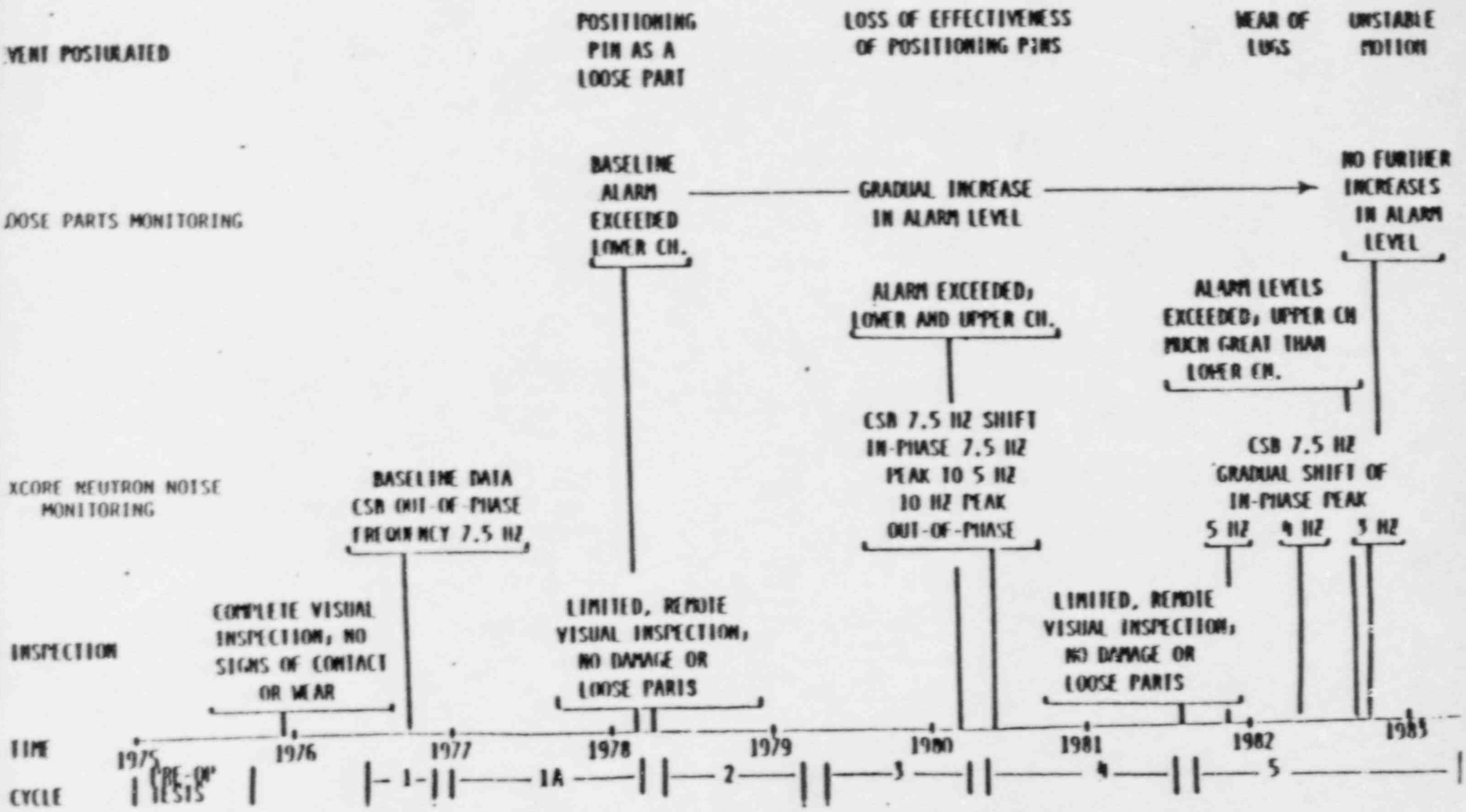

FIGURE 2

OMAHA CSB/TS GEOMETRY PLOT