APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report No. 50-482/92-06

Operating License No. NPF-42

Docket No. 50-482

Licensee: Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

Facility Name: Wolf Creek Generating Station (WCGS)

Inspection At: WCGS, Coffey County, Burlington, Kansas

Inspection Conducted: March 6 through April 1, 1992

Inspectors: D. R. Hunter, Senior Reactor Inspector, Operational Programs Section, Division of Reactor Safety

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Division of Reactor Safety, Region IV

Inspection Summary

Inspection Conducted March 6 through April 1, 1992 (Report 50-482/92-06)

Areas Inspected: Nonroutine, announced special inspection of the circumstances and activities associated with the noise event that occurred at the WCGS on February 28, 1992.

Results: Within the areas inspected, one non-cited violation was identified regarding a testing procedure deficiency. (paragraph 4)

The licensee's investigation of the noise events prior to the February 28 event was non-existent. Their method of investigation into the noise events

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following the February 28 event, however, was considered to be good. This investigation demonstrated appropriate involvement and integration of site, corporate engineering, and consultant personnel.

The licensee concluded that the noise events were caused by the release of the binding forces associated with the saddle block-to-support pipe whip restraints on the reactor coolant system piping. The licensee's conclusions were presented during technical meetings with the NRC staff and reviewed during the onsite inspection. As the result of these activities, the NRC considers the licensee's conclusions to be acceptable.

In addition to the non-cited violation, other weaknesses were identified. One weakness was the licensee's failure to investigate and provide corrective actions for the alarms and vibrations that occurred on January 9. Another weakness involved an inadequate procedure for the opening of the safety-injection (SI) accumulator isolation motor-operated valves. These two weaknesses are considered to be additional examples of problems with the licensee's self assessment and corrective action process and relate to a previous Enforcement Action (EA 91-161) in this area. A third weakness was identified involving a communications deficiency between the shift supervisory personnel and the operators with respect to the installed special instrumentation used to monitor for a thermal hydraulic event.

The following two inspection followup items were identified:

- Inspection Followup Item (482/9206-01): Review the licensee's final evaluation regarding the reactor coolant system (RCS) loop check valve testing activities. (paragraph 3)
- Inspection Followup Item (482/9206-02): Review the thermal growth monitoring procedure and the resulting data from the next two RCS heatups from below 440°F (paragraph 8)

DETAILS

1. EVENT SUMMARY

On February 28, 1992, the Wolf Creek Generating Station (WCGS) was conducting a routine reactor coolant system (RCS) heatup in preparation for a plant startup. At 1:54 a.m. a loud metallic sounding noise was heard and vibrations felt in the reactor containment building (containment). Following this occurrence, the licensee cooled down the RCS to cold shutdown (Mode 5) to investigate the cause of the noise.

The licensee notified the NRC of this event via a telephone call to Region IV on February 28. The NRC was informed that prior to this current event, the WCGS had experienced two additional noise events, one on May 10, 1990 and the other on January 9, 1992. During this telephone conversation the licensee discussed their plans to investigate the event and agreed to brief the NRC on the results of their investigation prior to taking the reactor critical. These actions were documented in a Confirmation of Action letter to NRC Region IV dated February 28, 1992.

The May 10 and January 9 events occurred as the plant was heating up following refueling outages. The February 28 event occurred as the plant was heating up following an unplanned reactor trip and scheduled maintenance outage. All three noise events exhibited the following similarities:

- The events occurred during the RCS heatup period prior to reaching normal operating temperature.
- A new procedure was used to establish a pressurizer bubble. This new procedure required a bubble to be formed at a higher RCS temperature (190°F vs. 130°F).
- All three events caused seismic alarms or seismic recorder actuation.
- All three events were witnessed or felt either in the control room, the auxiliary building, or the containment.

In addition, all three noise events exhibited the following differences:

- The three events all occurred at progressively escalating RCS temperatures and pressures as follows:
    * The event of May 10, 1990, occurred while the RCS pressure was being increased from 1100 to 1800 psig at a temperature of about 450°F.

* The event of January 9, 1992, occurred with RCS pressure at about 2000 psig and a temperature of about 500°F.

* The event of February 28, 1992, occurred with RCS pressure at about 2235 psig and a temperature of 537°F.

- The first event occurred prior to conducting RCS isolation check valve testing. The second and third events occurred within 6-9 hours after the completion of the check valve testing.

- The first two events occurred after recovery from reduced RCS inventory conditions. The third event occurred after a partial drain down (i.e., not mid-loop) to repair a leaking canopy seal on a vessel head penetration for a control rod drive mechanism.

- During plant startups the SI accumulators are required to be unisolated at an RCS pressure of approximately 1000 psig. During the first event, no system mechanical shocks were noted. During the second and third events, however, this evolution caused mechanical shocks to the RCS.

- Loose parts monitoring (LPM) system alarms did not annunciate during the first event. The LPM system alarms annunciated during the second and third events.

- A significant safety injection (SI) accumulator level decrease of about 33 percent occurred immediately following the first event.

Following the February 28 event, the licensee established an incident investigation team (IIT). The purpose of the IIT was to determine the cause of the noise events and recommend corrective actions. The initial IIT investigations took a three-path approach into investigating these events. The possibilities considered for the cause of the noise event were that: a part or component had become damaged or loose within the confines of the RCS or reactor vessel; a thermal-hydraulic event had occurred; or the system had been restrained from normal thermal expansion growth during the heatup. To assist in analyzing for the three possibilities, the licensee assembled technical expertise, which included contractors, consultants, and the services of problem solving experts.

The IIT established that the noises were not the result of a loose part within the RCS by satisfactorily completing system integrity inspections and analyzing LPM data. The IIT also verified that the core was intact by exercising all control rods and the core-flux mapping system in-core probes. This narrowed the noise events to either being caused by a thermal-hydraulic event caused as the result of RCS isolation check valve testing or a thermal expansion event caused by restrictions to RCS thermal growth.

Subsequent to the February 28 event and in preparation for a controlled plant heatup, so that sources of the noise could be determined, the licensee made plant procedure changes. In addition, to enhance their data collection capability, special temporary instrumentation was installed. To monitor for a thermal-hydraulic event, pressure monitoring instrumentation was installed on the RCS loops between the "first off" and "second off" check valves. In addition, the temperatures in these lines were monitored using thermography techniques. To monitor for the thermal expansion, five dynamic position transmitters were installed (one on each steam generator (SG) and one on the "B" reactor coolant pump (RCP) motor).

On March 9, a controlled plant heatup to normal operating temperature (557°F) was commenced. During this heatup personnel were stationed in the containment building and the heatup was halted at selected plateaus for data taking. At 7:20 p.m. on March 16, another loud noise was heard and vibrations were felt.

The plant status and observations associated with this event were as follows:

RCS temperature	551°F
RCS pressure	2235 psig
RCS loop check valve testing	Complete for 48 hours
Loose parts monitor alarms	Yes
Seismic alarms	Yes
Accumulator level change	No
Previous RCS drain down	None

Personnel stationed in the reactor containment reported that the noise sounded like a metal-to-metal impact and that they could feel vibrations.

The installed special instrumentation provided the following information:

- The pressure instrumentation installed on the check valve lines did not indicate a pressure change in the lines between the check valves.

- The thermography indicated that temperatures in the lines did not support steam formation or voiding.

- The thermal-growth monitoring instrumentation indicated that movement of three of the four RCS loops occurred.

Following investigations of this latest noise event by the IIT, the licensee concluded that the source of the noise events was due to restrictions in the thermal growth of the RCS during plant heatups. The licensee also concluded that binding was being caused by RCS crossover piping saddle blocks and supports (see ATTACHMENT C) that were installed as pipe whip restraints to mitigate a high energy line break. As the result of this conclusion, the licensee proceeded to correct the binding between the saddle blocks and supports. In addition, the licensee had determined that the stress levels on plant equipment and the support loadings and displacements for the affected systems did not exceed the design code allowable values. As a result of these determinations, this event was considered to have minor safety significance.

2. INITIAL INSPECTION ACTIVITIES

The inspectors arrived on the WCGS site on March 6. The immediate task undertaken by the inspectors was to interface with licensee's IIT and assimilate information regarding the event.

The licensee's IIT consisted of a team leader, five engineers, and an operations specialist. The licensee also obtained contractor assistance in the areas of piping and component inspection, piping stress analysis, interpretation of LPM and seismic-monitoring systems data, and failure cause determination. Later, during the investigation, the licensee obtained contractor assistance to evaluate the possibility of the occurrence of a thermal-hydraulic event.

The IIT had assembled data from the three identified noise events and selected documentation from the two outages that preceded the first event. This information was being used to conduct a change analysis to identify all possible similarities and differences that had occurred over the last five outages and subsequent plant restarts. The IIT was directed by administrative procedure, ADM 01-1'6, which provided the administrative aspects of event investigation such as event classification, charter development, team member appointment, preservation of evidence, preparation of an investigation plan, and report preparation. The procedure did not contain detailed guidance on root-cause analysis techniques or investigation methods; however, the licensee utilized failure-analysis technical experts in this area.

The inspectors began obtaining data, assembling an events and causal factor chart for each of the three known events, and interviewing licensee personnel. Each of the charts was constructed on a time line that commenced 24-36 hours before the event and ran until post-event, stable plant conditions were established. The timing and sequence of events was determined from control room logs, shift supervisor logs, event and alarm computer printouts, and the licensee's sequence of events for the three separate events. The purpose of this effort was to identify similarities and differences, among the three known events. The inspectors monitored, evaluated, and provided feedback to the licensee's investigation effort.

During interviews with licensee personnel, the inspectors determined that while the January 9 event resulted in both LPM and seismic monitoring system alarms and in vibrations being felt in the control room, the licensee failed to halt plant startup to investigate this unusual event. There was no attempt to assess the cause of this event nor to take corrective actions to assure plant integrity and prevent a recurrence. This inactivity by the licensee represents another example of an ongoing weakness with the licensee's self assessment and corrective action process that was identified in a previous Enforcement Action (EA 91-161).

The inspectors reviewed numerous design documents including piping layout drawings, piping isometric drawings, pipe-support drawings, system piping and instrumentation diagrams, and pipe-stress and pipe-support engineering data.

A partial listing of the documents reviewed is provided in Attachment B. This review examined the physical system configurations, the characteristics (predicted thermal and seismic movement) of the systems, and the mechanical condition of the systems relating to predicted stress levels, loadings, and displacements in the piping and support structures. The "B" RCS loop SI line was reviewed in detail because the licensee found a rotated pipe clamp on the line during system walkdowns after the February 28 event. The inspectors noted that the licensee had determined that the stress levels, and pipe-support loadings and displacements for the affected systems were within their design code allowable values.

The inspectors reviewed the results of the licensee's change analysis that was conducted shortly after the February 28 event. The analysis showed that the licensee had looked at changes occurring during the last five plant startups from cold shutdown conditions. The three broad areas selected for comparison were plant conditions, plant response, and unusual plant activities. The inspectors noted the licensee's conclusion that this change analysis did not reveal any possible causes of the events or suggest other areas to investigate.

The inspectors conducted walkdowns of the supports for the SGs and the RCPs for all four RCS loops to identify any visible damage or indications of abnormal movement that may have resulted from the February 28 noise event. A small scrape or gouge indication was found on the horizontal shim plate near the outer edge of the loop "D" crossover-leg saddle support on the SG side. Also, a vertical restraint (see ATTACHMENT C) attached to the "C" SG and anchored back to the concrete primary shield wall was found out of position with respect to a notch in a "C" SG support member. This notch allowed for thermal expansion of the restraint.

The Loop "B" 10-inch SI line running from the RCS cold leg out to the "B" accumulator tank was also examined. Signs of movement were limited to a rotated pipe clamp for snubber EP02-R010, which was located on the line adjacent to RCS "first off" isolation check valve, BB 8948B. A walkdown of the 10-inch SI line for the other three loops was also conducted with no signs of movement or damage noted.

3. <u>THERMAL HYDRAULIC EVENT INVESTIGATION</u>

The licensee's II investigation considered the possibility of a thermal-hydraulic event being the cause of the noise. This consideration was based upon the fact that steam-void formation might have occurred in the emergency core cooling system (ECCS). The potential existed that steam-void formation could have originated during the check valve testing process which required the area between the check valves to be depressurized, drained, and the flow measured to verify that the check valve leak rates were within licensed limits. The steam-void formation could have occurred within the area bounded by the RCS loop "first-off" check valves (valves BB 8948A, B, C, and D) and "second-off" check valves for the residual-heat removal (RHR) system (valves

EP 8818A, B, C, and D), SI system (valves EP V10, 20, 30, and 40), and SI accumulators (valves EP 8956A, B, C, and D) (See ATTACHMENT D).

As the result of this potential, the licensee decided to instrument the portion of the ECCS system where steam-void formation could occur so that pressure could be recorded. To implement this change, a temporary change to ECCS check valve testing procedure, STS PE-019E, was initiated to require pressure monitoring of the ECCS system area bounded by the "first-off" and "second-off" check valves. The change also required temperature monitoring by thermography across the RCS isolation check valves to determine if leakage back through the check valves and/or steam-void formation or collapse was occurring. Check valve testing was to be integrated with procedure TP-IS-72.

The inspectors reviewed the conditions, procedures, drawings, and practices established by the licensee to perform the RCS isolation check valve leak-rate testing. Document reviews and personnel interviews revealed that the basic testing methodology established in the test procedure appeared to provide acceptable leak-rate flow information. The review of the system piping configurations revealed that any leakage into the SI headers or test lines from other sources would be indicated conservatively as RCS isolation check valve leakage.

A review of the leak-rate testing data, recorded on February 27 for the "first-off" RCS isolation check valves, by the inspectors indicated that between 50 and 200 gallons of liquid were removed from each 10-inch SI cold-leg header during the testing. At the completion of the testing of each of these "first-off" check valves, the associated SI accumulator isolation motor-operated valve was opened, and the associated SI accumulator level decreased between 7 and 20 percent (56 to 160 gallons), indicating a transfer of liquid to refill the partially drained 10-inch SI cold-leg headers.

The inspectors determined that the 10-inch cold-leg SI headers, depending on the RHR systems in operation (RHR Train A returns to RCS loops 1 and 2, RHR Train B returns to RCS loops 3 and 4) would be at or near the RCS temperatures. Additionally, subsequent to securing the RHR systems during normal plant heatup activities, the 10-inch SI headers remained at an elevated temperature of about 325°F, depending on the leakage of reactor coolant through the individual "first-off" 10-inch check valves and the "second-off" 6-inch check valves into the idled RHR systems. While observing the local opening of the accumulator isolation motor-operated valves on March 11, 1992, the inspectors noted that the temperatures of the 10-inch SI headers were not the same. The RCS loop "first-off" check valve leakage appeared to be higher at lower RCS pressures due to a reduced differential pressure across the check valves. As a result of the previously experienced RCS coolant leakage through the check valves into the idle RHR system, the licensee had established an off-normal procedure, OFN 00-028, to address these specific conditions. The inspectors observed above-ambient temperatures on the RHR systems and the implementation of the off-normal procedure on March 11 and 12.

The inspectors reviewed the RCS isolation check valve test data for the testing conducted on March 14. The data documented a stable leakage flow rate within about 30 minutes for each check valve. Personnel interviews and document reviews revealed that the normal testing sequence included the use of the local hydrostatic test pump with a discharge pressure of about 1750 psig to ensure the cold and hot leg headers were filled and pressurized. While it appeared that this technique was effective, the inspectors noted that the cold and hot leg SI headers were not necessarily maintained filled and pressurized in all cases. The pressures in the SI headers were not routinely monitored to ensure the headers were maintained filled and pressurized. As a result, limited RCS leakage into the low pressure standby RHR or SI systems had been noted as indicated by elevated RHR or SI temperatures and pressures.

The noted variations in SI header temperatures and pressures, the recorded level changes in the SI accumulator tanks, and other routine operations (e.g., operations of the SI pump to fill accumulators) were discussed with the licensee. The licensee was continuing to review and evaluate the RCS isolation check valve testing activities and parameters as a part of their event investigation.

Since these activities were not completed prior to the end of this inspection, the licensee's final evaluation of the check valve testing activities will be examined during a future inspection. (Inspection Followup Item 482/9206-01)

On March 12, the inspectors attended a briefing conducted for personnel who would be taking data and performing local operations during check valve testing. The briefing was thorough, detailed, and addressed all concerns of those personnel present. The installation and calibration of the special instrumentation was observed by the inspectors. The inspectors noted that while the shift supervisory personnel were aware of the installation and purpose of the temporary instrumentation, the control room panel operators were not aware that the temporary modification associated with the special instrumentation had been installed. The inspectors concluded that either the testing program or the temporary modification program had not functioned to train or inform all of the control room operators of the special testing and the resultant ECCS test configuration. Licensee management was informed of this apparent communication weakness.

4. SI ACCUMULATOR ISOLATION MOTOR-OPERATED VALVE CONTROL

During the inspection, one of the issues reviewed by the inspectors was the apparent improper operation of the motor-operated valves used to isolate the SI accumulators during plant cold shutdown conditions. The licensee had identified that the "A" 10-inch SI cold-leg header was subjected to a mechanical shock on January 8, 1992, when the "A" accumulator isolation motor-operated valve was opened. The inspectors reviewed the maintenance history associated with the 10-inch SI headers (valves, supports, hangers, etc.,). The records review and personnel interviews revealed that corrective work requests were written to realign rotated pipe clamps on the "A" and "C" 10-inch SI lines on two occasions in January 1992. However, the corrective actions did not include a thorough investigation of the condition to determine the cause of the rotated pipe clamps (e.g., mechanical shocks) and provide corrective actions to prevent a recurrence (e.g., through procedure changes and training). During subsequent routine surveillance testing activities associated with the RCS isolation check valves on February 27, 1992, the licensee identified that the "A," "C," and "D" 10-inch cold leg headers were again subjected to mechanical shocks, when opening the associated accumulator isolation valves. The mechanical shock which occurred on the "D" header resulted in the annunciation of both LPM and seismic-monitor alarms.

As the result of a review of these occurrences, the licensee determined that if the SI accumulator isolation valves were opened slowly to 1/4 to 3/8 of full open, the mechanical shock to the SI piping and supports could be prevented. Interviews with licensee personnel revealed that cracking open of the valve slowly, to equalize pressure, then slowly opening the valves to 100 turns was considered to be an acceptable practice that would ensure that the fluid velocities were limited while refilling the 10-inch SI header thus preventing the mechanical shocks. As a result, test procedure STS PE-19E was subsequently changed to provide this method for opening the valves.

Procedure GEN-00-002 required that the SI accumulator isolation motor-operated valves were to be opened at a RCS pressure greater than 950 psig and then that the electrical circuit breakers associated with each individual valve were to be opened. The inspectors observed the opening of these SI accumulator isolation valves in accordance with GEN-00-002 by operating personnel on March 11 with the RCS pressure about 975 psig. During this specific evolution, the procedure also required the shift supervisor to verbally provide additional guidance to the operators regarding the method of opening of the valves. The guidance was that each valve be operated locally by slowly cracking open the valve and then opening the valve 100 turns using the handwheel. The valves were then closed electrically by the operator in the control room.

Although the verbal guidance was acceptable, the inspectors were concerned about the lack of specific procedure details to ensure acceptable and consistent operation of these valves. This issue was of concern because of the potential for resultant mechanical shocks and was brought to the attention of the licensee. The licensee indicated that the procedure steps associated with the opening of these valves would be changed to reflect the methodology established to prevent mechanical shocks.

While the licensee had recognized the problem with the operation of the SI accumulator isolation valves and had taken corrective action with a revision to the testing procedure (STS PE-19E), they had failed to recognize that a change to procedure GEN-00-002 was also required. This weakness is considered as an additional example of the failure to provide thorough and timely corrective actions to an identified condition. The inspectors considered this matter related to the ongoing overall upgrade of the WCGS corrective-action process resulting from a previous Enforcement Action (EA 91-161).

The licensee is required by Technical Specification (TS) 6.8.1, section 8.b, to have adequate written surveillance procedures. Surveillance test procedure STS PE-19E was inadequate in that it did not provide appropriate guidance regarding the method of opening the SI accumulator isolation motor-operated valves to assure that the RCS was not subjected to damage due to mechanical shocks. Such inadequacies are considered to be a violation of the plant's license conditions. This violation was discovered by the licensee and was corrected by the licensee prior to the completion of this inspection. Surveillance test procedure STS PE-19E was revised and personnel were trained in the use of this procedure. This violation is not being cited because the criteria specified in Section VII.B(2) of the Enforcement Policy were satisfied.

5. THERMAL EXPANSION EVENT INVESTIGATION

Subsequent to the February 28 noise event, the licensee's quality control and engineering staffs had performed numerous walkdowns in the containment to determine if any piping or support irregularities had occurred. Walkdowns were conducted on the RCS, ECCS, and main feedwater piping. When the inspectors arrived on site, they were informed by the licensee's IIT that they considered the following items as possible indications of the noise events: (Refer to ATTACHMENT C)

- The "D" loop RCP end crossover-leg saddle support exhibited a 4-inch by 1/2-inch gouge on the shim plate surface.

- A small spring was found on the shim plate surface of the "C" loop RCP end saddle support.

- The "first off" pipe support on the "B" RCS loop accumulator SI line (support EP02-R010) had a rotated clamp.

As the result of insufficient information to determine the root cause for any of the noise events, the licensee made a decision to monitor the RCS thermal-expansion growth during stringently controlled RCS heatup conditions. On March 5, 1992, the licensee began preparing test procedure TP-TS-72 to provide a controlled method for monitoring the thermal expansion of the selected RCS components. The components were selected based upon discussions with the reactor vendor and the architect engineer. The procedure specified data plateaus of approximately 125°F, before and after drawing a bubble in the pressurizer at 190°F, 350°F, 425°F to 450°F (corresponding with check valve testing), and 557°F (no-load normal operating temperature). At each plateau the plant was stabilized for 1 hour prior to taking data.

The procedure addressed the following monitoring activities:

- Selected snubbers were free of interference or obstruction, and exhibiting normal support characteristics by measurement.

- Selected pipe whip restraints exhibited proper gap clearance initially and during heatup.

- Selected spring supports were verified to be within the operating range.

- RCP tie-rod supports were measured for proper motion.

- Crossover-leg out-of-plane motion was continually measured and recorded by dynamic-position transmitters.

- Crossover-leg saddle block and support horizontal and vertical gaps were measured.

The procedure required all data taken at each temperature plateau to be analyzed prior to proceeding to the next higher plateau.

The inspectors reviewed the approved procedure and the engineering safety evaluation, and the matter was discussed with licensee personnel. The inspectors concluded that the monitoring plan was sufficiently detailed to either eliminate or confirm that thermal-expansion growth of the RCS or connected systems was causing the noise anomaly.

On March 9, the heatup was commenced, monitoring was performed, and data analyzed up through the 450°F RCS temperature plateau. Upon reducing and performing analysis of the data and inspection observations, the licensee identified the following anomalies:

- There was a questionable setting on a pressurizer relief valve discharge line spring support.

- Crossover-leg saddle block to support gaps appeared to be inadequate thereby allowing the blocks and supports to come in contact at an RCS temperature of approximately 520°F. This condition could cause binding of the sliding surfaces.

- The clamps for the cold-leg crossover pipe vertical restraint that was clamped to the RCS piping had slipped down as much as 6 inches on three loops.

The first two items were entered into the licensee's corrective action system in the form of corrective work requests to adjust or repair at a later time, however, the licensee did verify that any stress buildup that could occur due to saddle block-to-support contact at 520°F, were well within the design code allowable values. Action was taken to correct the mispositioned vertical restraints.

The licensee initiated a documentation search to ascertain the gap clearance recorded during the 1984 facility hot-functional testing conducted during final plant construction and prior to initial fuel load. The hot-functional testing documentation indicated that the crossover-leg out-of-plane, vertical, and horizontal growth had been measured during heatup with the support shims removed. Once the thermal growth had been determined, shim plates were machined and installed to provide for the proper saddle block-to-support gap clearance at normal operating temperatures. The documentation did not indicate that the proper gap clearance existed following the hot-functional testing.

The plant heatup was resumed from the 450°F plateau and at 7:20 p.m. on March 16 the fourth noise event occurred. The licensee analyzed all the data that had been collected since the heatup had recommenced. Preliminary investigation and analysis of the data indicated that RCS loop thermal expansion out-of-plane growth had been restrained during the heatup by at least one of the crossover-leg saddle blocks coming into contact with its respective support. When the out-of-plane growth force overcame the friction forces caused by the deadweight loading of the saddle blocks on the supports, the blocks moved suddenly, providing stress relief, and resulting in the loud noise. Additional analysis of the data collected during the final stages of the RCS heatup confirmed that insufficient gap clearance between the saddle blocks and their supports was the most likely cause of this event. The licensee also concluded that this was the cause of the noise from the two most recent events.

Following the noise event, the licensee cooled the RCS down to 440°F to evaluate the condition of the eight crossover-leg saddle blocks and supports. The licensee's initial inspection revealed that some of the saddle blocks had been in hard contact with the supports in the shim plate area. There was also indication that there was some debris found in the gap area and that this debris may have interfered with a smooth sliding contact surface between the horizontal surface of the saddle blocks and shim plates. Further, there appeared to be minor deformation of some of the horizontal shim plates. The licensee's inspection also included a check to determine if the clamps holding the saddle block to the pipe elbow had slipped. No such slippage was observed.

A decision was made to remove all eight (two per loop) horizontal shim plates and the RCP saddle support vertical shim plates (four total). It was also decided to leave the SG end crossover-leg saddle support vertical shim plates as-is because the RCS loop support system was designed for longitudinal expansion of the crossover-leg toward the RCP and, therefore, clearance only needed to be assured at the RCP end. Quality control documents revealed that the following conditions had been identified by the licensee upon shim plate removal:

- Loop "A" saddle supports for both the SG and RCP had raised metal areas on the horizontal and vertical shim surfaces. There were six pieces of shim stock stacked underneath the SG end horizontal shim plate.

- The loop "B" SG saddle block had raised metal on the horizontal surface. There were also pieces of shim stock under one corner of the shim plate. A weld fillet gauge was found under the RCP end saddle support horizontal shim plate.

- On loop "C", a spring had been pressed into the SG end saddle support horizontal shim plate upper surface and had left a 1/16-inch deep impression. There was raised metal on both horizontal saddle blocks.

- The "D" loop SG end saddle support had three pieces of shim stock "massed" under the shim plate. A nail was also embedded in the horizontal shim plate which contained a 3-inch x 1/16-inch gouge near the "massed" shim stock.

The purpose for the saddle block and support shim plates removal was to restore the clearance gaps to design tolerances by machining the surfaces. The plan was to use procedure TP-TS-73 to monitor system thermal growth and support performance during the RCS heatup. The resultant measured thermal expansion displacement would provide the data necessary to restore the saddle block-to-support clearance gaps to the design conditions.

Each of the 12 shim plates was modified and reinstalled under separate work packages. The inspectors observed the reinstallation work in progress and reviewed selected parts of the working and completed packages. The work procedures required that shim plates be installed and adjusted with shim stock to obtain a uniform gap of 1/16 inch +/- 1/32 inch with the RCS at normal operating temperatures. An absolute parallel gap was not totally achievable for all restraints because of previous plate deformations. However, a good degree of parallelism was assured by setting the specified gap at each corner and the center of each block and plate interface. The inspectors observed full-time job-site quality control coverage for the reinstallation work. To minimize further shim plate deformation from the welding process, the weld procedure required a metal preheat to 350°F. Some of the completed packages identified final gap clearances that were as much as 0.010 inches over or under the specified gap. These conditions were reviewed and accepted by the licensee's engineering staff.

6. LOOSE PARTS MONITORING AND SEISMIC MONITORING SYSTEM INVESTIGATION

During the initial investigative period, the licensee was evaluating the data that had been retrieved from the seismic monitoring and the LPM systems. All data that was available from the January 9 and February 28 events was collected for the preliminary analysis. The results of this initial analysis concluded that the possibility of a loose part or component being responsible for the noise was remote. However, analysis in this area continued as the available data was being enhanced by a reduction to shorter time intervals. The licensee was performing an evaluation regarding the latest two events (i.e., February and March), to make a comparison of various seismic triaxial accelerometer values and to further compare the available seismic data to the design limits.

The inspectors reviewed data from the LPM system and the seismic instrumentation system in an attempt to determine the location where the noise event originated. The LPM sensors were located at various points throughout the RCS to detect loose parts or abnormal vibration. The seismic instrumentation system provided information on the effects of seismic events through the use of a combination of strong motion accelerometers, peak reading accelerographs, and seismic triggers and switches located throughout the power block with a free-field accelerometer located outside of the power block. (see ATTACHMENT E)

A review of the February 28 event data indicated that the event was of a very short duration and low response; motion was predominately in the east-west direction; and the event most likely originated in the vicinity of the "D" SG. This conclusion was based upon the initiation sequence of the LPM system sensors and the activation and sequence of the seismic monitoring system devices. The LPM sensors on the "D" SG were the first to signal, followed by the sensors located at the bottom of the reactor pressure vessel, with the "A" SG sensors next, and the remaining sensors following. The response of the seismic monitoring system to the noise event also indicated that the point of origin was in the vicinity of the "D" loop. Of the six accelerometers used in the system, only three were activated by the noise event. The first to activate was SG-AE-3 located on the primary shield wall at elevation 2013 feet, followed by SG-AE-1 located at the auxiliary building basemat (elevation 2000 feet, where the containment building abuts the auxiliary building) and finally SG-AE-2 located at elevation 2056 feet on the containment structure (Refer to ATTACHMENT E). All three of these devices were located near azimuth 270° of the containment building with the "D" SG device located less than 12 feet from azimuth 270°. In addition, the free-field accelerometer device did not activate during the noise event, indicating the origin of the event was not seismic in nature, but originated inside the containment building structure.

On March 7, 1992, the inspectors attended a licensee briefing on the seismic data analysis conducted by a seismic monitoring expert. From review of the seismic monitor data, the expert determined that the event lasted 0.45 seconds and that the largest forces and displacement occurred in the east-to-west direction at elevation 2013 feet. The focus and displacement indicated that the forces were localized. In summary, the seismic expert determined that the event occurred over a short period of time, was localized, had a low response, and was not continuous. The licensee concluded that the event was of insufficient duration to input significant damage causing energy into the structures.

7. REPORTABILITY

The licensee had made a determination that the February 28, 1992, event was not reportable in accordance with Procedure ADM 01-033, "Reportability Evaluation Request." The inspectors reviewed the licensee's reportability procedure against 10 CFR Part 50.73 reportability criteria and determined that the licensee's procedure incorporated the requirements of the regulation. The licensee's documentation indicated that the event had been considered potentially reportable under 10 CFR Part 50.73, Sections (a)(2)(ii) or (a)(2)(v). However, when initial walkdowns and reviews indicated that no system or plant degradation had occurred, that no condition was identified as outside the design basis, that the facility was in an analyzed condition, and that safety systems would function as designed, the licensee concluded there was no basis for reportability pursuant to 10 CFR Part 50.72 or 10 CFR Part 50.73. The inspectors concluded that neither regulations nor the licensee's procedures specifically required the event to be reported. The licensee notified the NRC senior resident inspector on February 28, 1992, of that event. The licensee also planned to provide a voluntary written report (LER 92-006) to the NRC in the near future.

8. LICENSEE CORRECTIVE ACTIONS AND COMMITMENTS

During meetings with licensee management personnel, plans for short and long term corrective actions were discussed with the inspectors. The following revisions to some key procedures had been initiated during the inspection period:

- Procedure GEN 00-002 was revised to require that accumulator isolation valves be slowly cracked opened locally in manual to minimize the potential for the mechanical shocks that had been experienced during previous startups.
- Procedure STS PE-019E was revised to require the same method of opening the accumulator isolation valves as above, whenever check valve testing required these valves to be operated. This change was intended to minimize the potential for the mechanical shocks.
- Annunciator Response Procedures ALR 00-98E, "Seismic Recorder On," and ALR 00-98C, "R Spectrum OBE Exceeded," were revised to require the permission of the Director of Operations or the Call Superintendent to change plant conditions if these alarms annunciated.

The inspectors reviewed the changes and had no further questions.

The licensee stated that it was their intention to monitor RCS thermal growth for the next two plant heatup evolutions from below 440°F utilizing a procedure similar to the thermal-growth monitoring procedure, TP-TS-73. In further discussions, licensee management committed to develop acceptance criteria for the monitoring procedure, which would initiate an investigation, evaluation, and implement corrective actions in the event deviations from expected conditions were detected during the heatup. The monitoring procedure and resulting data from the next two plant heatups will be evaluated during a subsequent inspection. (Inspection Followup Item 482/9206-02)

9. PUBLIC MEETINGS

Two meetings were conducted between the licensee and the NRC staff on March 24 and March 26, 1992. The meetings were open to the public.

The March 24 meeting was held at NRC headquarters in Rockville, Maryland. During this meeting, the licensee presented the results of their IIT investigations including their engineering analysis and testing methodology. The presentation provided the engineering basis which eliminated the loose parts concern within the RCS or a thermal-hydraulic event as possible causes of the noise. The presentation also demonstrated that insufficient clearance between the saddle block and shim plate of at least two crossover-leg pipe whip restraints on different loops had restrained RCS thermal growth. The subsequent sudden release of the energy restraining the thermal growth had caused the noise for the last three events. A root cause for this insufficient clearance had not been determined; however, possible contributing causes were stated to be:

- Deformation of the restraint shim plates caused by the installation welding process.
- Deformation of the shim plates caused by previous deadweight loading during past heatups.
- Degradation of the surfaces on the saddle blocks and matching shim plates because of a cold-weld phenomenon resulting from deadweight loading.
- Degradation of the sliding surfaces as a result of the presence of debris between the shim plates and blocks.

In addition, information was provided by the licensee to demonstrate that stress analyses had been performed to verify RCS integrity. The licensee had bounded "as-found" conditions and performed a number of evaluations, which included piece-wise linear-thermal expansion, fatigue, and snapping analyses. Further, the resultant primary equipment nozzle loads and predicted accumulator hydraulic loads that occurred during check valve testing had been evaluated. Finally, the stresses associated with the predicted RCS over-temperature transients were addressed. While the facility had not previously experienced any over-temperature transients, the conditions encountered were bounded by the over-temperature transient analysis. These stress analysis packages were subsequently reviewed by the NRC staff. The staff determined that the licensee's conclusions were valid. A meeting summary was issued by the NRC on April 15, 1992.

On March 26, a second meeting was conducted at the licensee's Dwight D. Eisenhower Training and Education Center. This meeting was held to describe the noise event investigation results and to present the specific corrective actions taken including the long-term monitoring and action plan. Following this meeting, the licensee was given NRC concurrence to resume plant operation.

A copy of the licensee's presentation, which was provided as a meeting handout, is included in the report as ATTACHMENT F.

10. CONCLUSIONS

The licensee's detailed investigation of the noise events was considered to be good. Corporate engineering and consultants were appropriately utilized by the licensee to address the apparent causes of the noise events, and to ensure that the plant was in a safe condition throughout the investigation.

Based on the inspectors' walkdowns, inspections, and review of the three most recent noise events, it was concluded that the restraint of system thermal growth caused by binding of the RCS crossover-leg saddle block and support pipe whip restraints, and the subsequent energy release of the restraining forces, was the most probable cause for the events. The information from special monitoring systems that were in place when the noise event occurred again on March 16 confirmed the existence of thermal expansion restraints caused by the binding of the RCS "B," "C," and "D" loops.

An instance was identified, regarding the installation of the special test instrumentation, where the operator knowledge could have been improved. This indicated a weakness in communications between shift supervisory personnel and the operators.

The following three concerns were identified where thorough and timely corrective actions were not provided:

- The licensee failed to pursue the source of the noise that occurred on January 9. Even though the noise resulted in both LPM and seismic monitor system alarms and caused vibrations that were felt in the control room, no action was taken to identify the cause of the alarms or the vibrations.

- The licensee failed to evaluate the root cause of the mechanical shocks on the RCS loops that occurred in January 1992 when the SI accumulator isolation motor-operated valves were opened.

- The licensee failed to make the necessary changes to all of the appropriate procedures associated with the routine opening of the SI accumulator isolation valves to assure that further mechanical shocks to the RCS would be minimized.

These concerns demonstrate a weakness with the licensee's self assessment and corrective action activities and are considered to be additional examples of the issues identified in Enforcement Action (EA) 91-161.

10. EXIT MEETING

An exit meeting was held at the conclusion of the inspection on April 1, 1992, with the personnel denoted in Attachment A. The public was invited to attend this meeting. The inspectors summarized the scope and findings of the inspection. The licensee did not identify, as proprietary, any information provided to the inspectors during the inspection.

ATTACHMENT A

PERSONS CONTACTED

WCNOC

*J. Bailey, Vice President, Nuclear Operations
B. Du[illegible]p, Supervising Operator
R. Ev[illegible]on, Supervising Operator
*R. Fla[illegible]igan, Manager, Nuclear Safety Engineering
*R. Hagan, Director Nuclear Services
*F. Hall, Supervisor, Quality Control
*T. Hood, Supervising Engineer, Plant Design Engineering
*K. Hughes, Supervisor, Training Development
R. Hoyt, Operations Support
J. Isch, Engineer, Results Engineering
*R. Lewis, Supervisor, Results Engineering
*W. Lindsay, Manager, QA
*O. Maynard. Director, Plant Operations
B. McKinney, Manager, Training (Outage Manager)
R. Miller, Shift supervisor
K. Montgomery, Engineer Results Engineering
D. Moseley, Operations Supervisor
L. Parmenter, Supervising Operator
*C. Parry, Director, Quality and Safety
G. Pendergrass, Supervisor, Results Engineering
*J. Pippin, Director, Nuclear Plant Engineering
R. Reitman, Lead Engineer, Engineering Results
*F. Rhodes, Vice President, Engineering and Technical Services
R. Richardson, Control Room Operator
*A. Riezer, Engineer, Nuclear Safety Engineering
R. Schneider, Shift supervisor
*R. Smith, Information Administrator
*J. Stamm, Manager, Plant Design Engineering
*L. Sterns, Supervising Engineer, Nuclear Safety Engineering
D. Walsh, Reactor Operator
T. Warner, Reactor Operator
*J. Weeks, Manager, Operations
*S. Wideman, Supervisor, Licensing
*B. Withers, President and Chief Executive Officer
*W. Wood, General Counsel

Others

*G. Allen, Kansas Department of Health and Environment
*E. Creel, Manager, Nuclear Activities - Kansas Gas and Electric
*W. Goshorn, Kansas Electric Power Company (Wolf Creek Coordinator)
*C. Ross, Director, Power Production Engineering - Kansas City Power and Light

NRC

*T. Stetka, Chief, Operational Programs Section, Region IV
*G. Pick, Senior Resident Inspector
B. Bartlett, Senior Resident Inspector, Callaway

Other members of the operational, technical, and administrative staffs were contacted during the inspection.

*Denotes those present at the exit meeting held on April 1, 1992.

ATTACHMENT B

Documents Reviewed

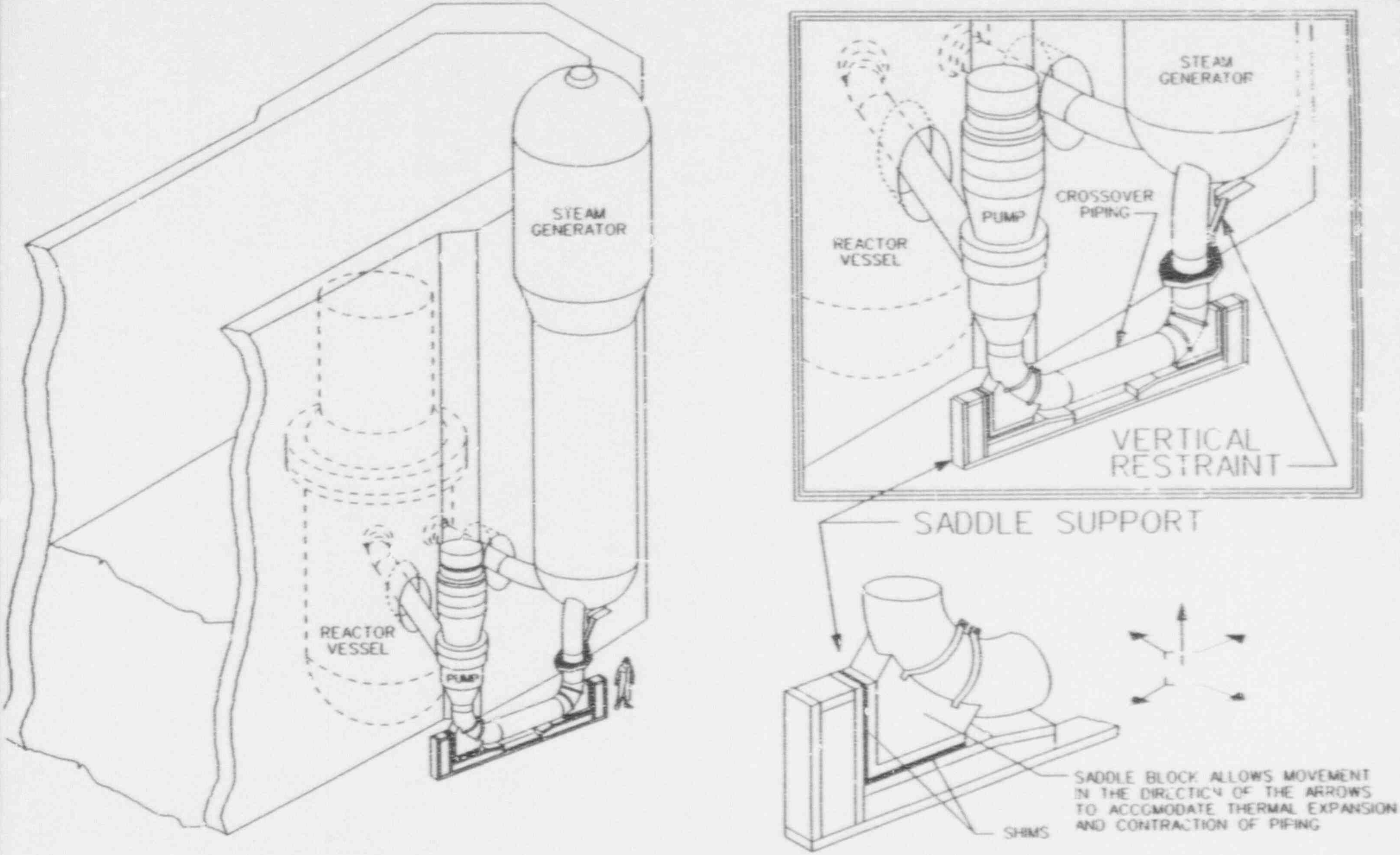
Procedures

ADM 01-116, "Incident Investigation," Revision 4
GEN 00-002, "Cold Shutdown to Hot Standby," Revision 22
SYS EP-201, "Seating, Accumulator Safety Injection Line Check Valves Using Test Line Valves," Revision 1
OFN 00-028, "Idle RHR Train Temperature Control Modes 1-4," Revision 3
STS PE-19E, "RCS Isolation Check Valve Leak Test," Revisions 4, 6, 7, and 8
TP-TS-72, "Thermal Expansion Monitoring Procedure," and Revisions 1 through 5
TP-TS-73, "Thermal Growth Monitoring Procedure," Original
STS PE-040B, "Reactor Coolant System Pressure Test," Revision 2
STS EJ-001, "Containment Inspection ," Revision 4

Drawings

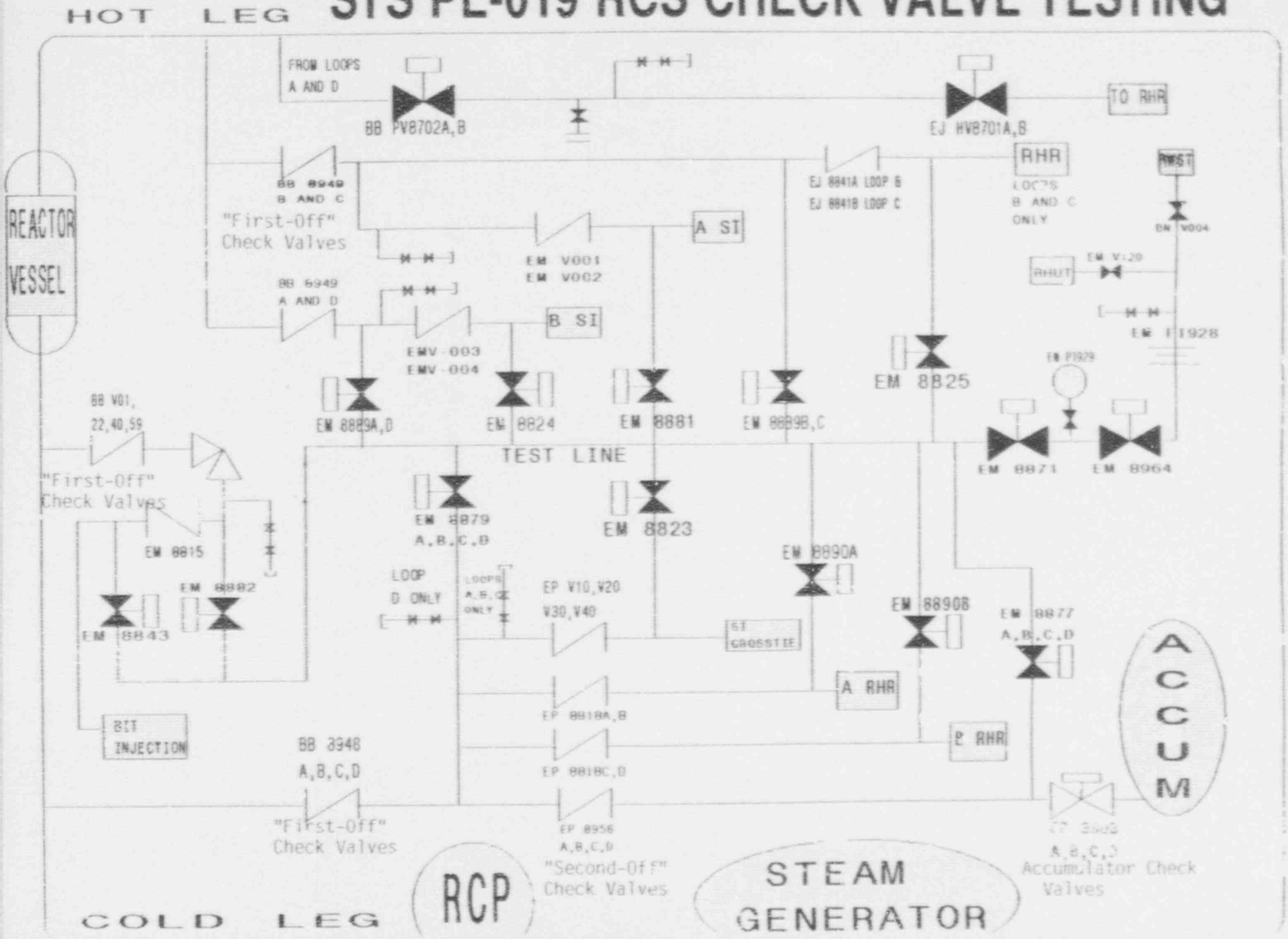
M-12EJ01 (Q), Residual Heat Removal System, Revision 1
M-12EP01(Q), Accumulator Safety Injection, Revision 0
M-13EP01(Q), Accumulator Safety Injection System Loop No. 1 and 4, Revision 3
M-13EP02(Q), Accumulator Safety Injection System Loop No. 2 and 3, Revision 9
M-13EM03(Q), High Pressure Coolant Injection system, Revision 6
M-13BB01(Q), Reactor Coolant System, Revision 4
M-12BB01(Q), Reactor Coolant System, Revision 4
M-12BB01(Q), Reactor Coolant System, Revision 4
M-12EM01(Q), High Pressure Coolant Injection System, Revision 2
M-12EM02(Q), High Pressure Coolant Injection system, Revision 2

ATTACHMENT C

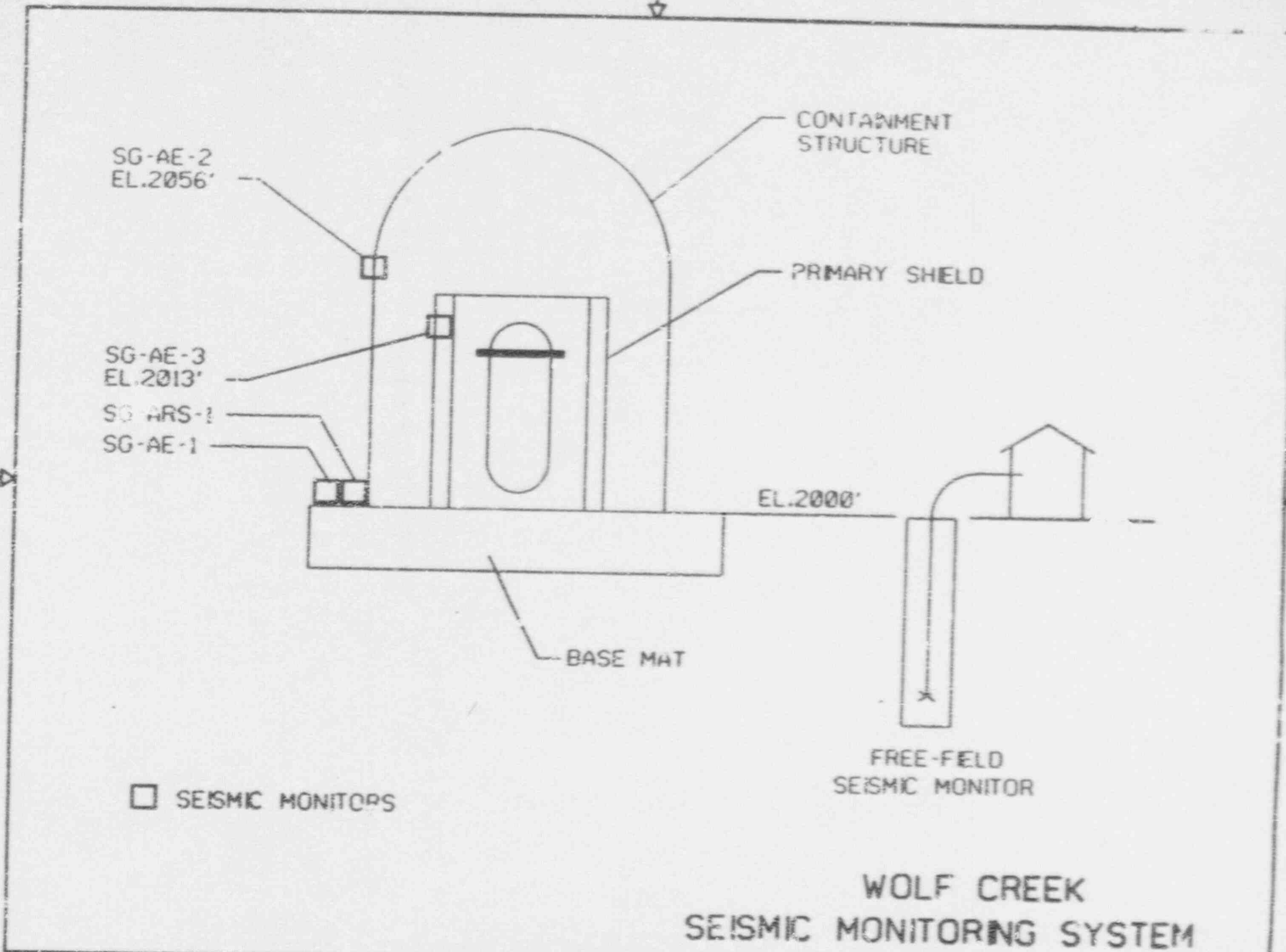


ATTACHMENT D

STS PE-019 RCS CHECK VALVE TESTING



ATTACHMENT E



WOLF CREEK
SEISMIC MONITORING SYSTEM

ATTACHMENT F

Meeting on Recent Noise Events

March 26, 1992

WOLF CREEK
NUCLEAR OPERATING CORPORATION

Agenda

Introduction	Bart Withers
I. Objectives of Meeting	Otto Maynard
II. Noise Event of February 28 - Initial Actions and Investigation	Otto Maynard
III. Description of the WCNOC Investigation	Rich Flannigan
Loose Parts	Brad Norton
Thermal-Hydraulic Mechanisms	Tom Hood
Thermal Growth	John Stamm
IV. Identification and Correction of the Cause	John Stamm
V. Safety Impact and Plant Operability	Otto Maynard
VI. Summary	Bart Withers

I. Objectives of Meeting

Purpose of the meeting:

- demonstrate that noise events caused no damage
- demonstrate cause of noise has been corrected
- demonstrate that Wolf Creek is ready to safely resume operation

The presentation will describe the following:

- the noise events and immediate actions;
- the investigation that has been undertaken;
- the identification and correction of the cause;
- the results of safety impact and operability analyses; and
- anticipated longer term actions

II. The Noise Event of February 28 and Immediate Actions

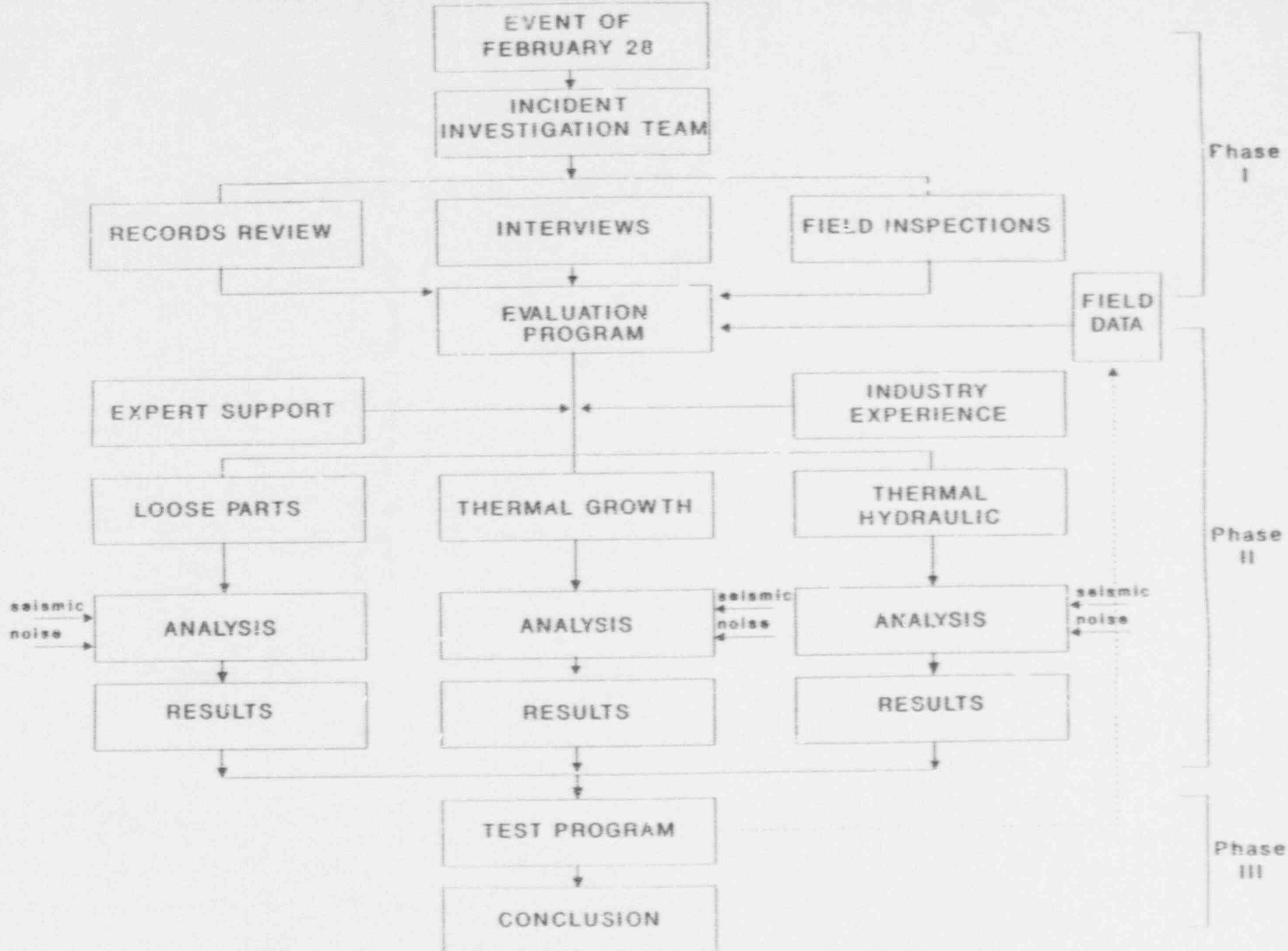
- During conduct of containment walkdown at pressure for a reactor mode change
- Noted canopy seal weld weepage and commenced an in-plant evaluation
- While conducting seal weld evaluation, the noise occurred with accompanying control room alarms
- Held plant conditions constant and conducted containment walkdown to look for obvious damage
- Evaluated Reportability
- Conducted the following actions in the morning of the first day
  - Initiated a company Incident Investigation Team (IIT) review
  - Began on-going information exchange with NRC Region IV
    Sent letter to RIV committing to keep them informed of our intentions
  - Sent Loose Parts Monitor Tapes off for analysis
  - Checked reactor coolant system leakage rate
  - Began a detailed containment walkdown - confirmed ECCS availability

II. The Noise Event of February 28 and Immediate Actions

(continued)

- Without immediate discovery of a cause or apparent effects, went to cold shutdown and conducted an additional walkdown
- Began root cause investigation using change analysis methodology

WOLF CREEK NUCLEAR OPERATING CORPORATION
FEBRUARY 28 NOISE EVENT INVESTIGATION OVERVIEW



III. Description of WCNOC Investigation

The investigation involved in-house expertise supplemented by outside experts. Contributions were as follows:

- WCNOC Staff - approximately 125 persons - engineering, maintenance, instrument and control, operations, health physics, quality control, quality assurance, and management

- Westinghouse - provided assistance with: piping stress analysis, bounding calculations; loose parts noise analysis; and safety evaluation

- Bechtel - provided assistance with: seismic analysis; pipe load analysis; and thermal-hydraulic investigation

- MIT - provided assistance with thermal-hydraulic modeling

- Failure Prevention, Inc. - provides assistance with root cause analysis

III. Description of the WCNOC Investigation
(continued)

The IIT used a variety of means to gather information including:

- records review
- interviews
- field inspections
- major modifications
- industry experience
- RCS check valve testing

Within 48 hours, identified the event of January 9 and determined that it was similar to the February event - began to focus on commonality

Used coarse screening criteria (unexplained control room seismic alarm and confirming report of personnel in control room logs) to identify other possibly similar events

Attributes of Noise Events

Date	Witnessed/ Felt*	Accumulator Isolation Valve/Level Deviation	Loose Parts Alarm	Seismic Alarms* 98C	98E	Accumulator Check Valve Testing	Engdahl Seismic Inst.	Temperature Pressure		B RCP Vibration
May 1990	Yes	Open/Yes	No	No	Yes	Testing in progress	Not Available	Steady Rising	450°F 1100 psig	No
Jan 1992	Yes	Open/No	Yes	Yes	Yes	Complete (6 hrs)	32 hz E-W	Rising	512°F 1830 psig	Yes
Feb 1992	Yes	Open/No	Yes	Yes	Yes	Complete (9 hrs)	32 hz E-W	Steady	537°F 2235 psig	Yes
Mar 1992	Yes	Open/No	Yes	Yes	Yes	Complete (48 hrs)	32 hz E-W	Rising	551°F 2235 psig	Yes

The dissimilarity of the May 1990 to the other three events lead to the conclusion that the event is not related.

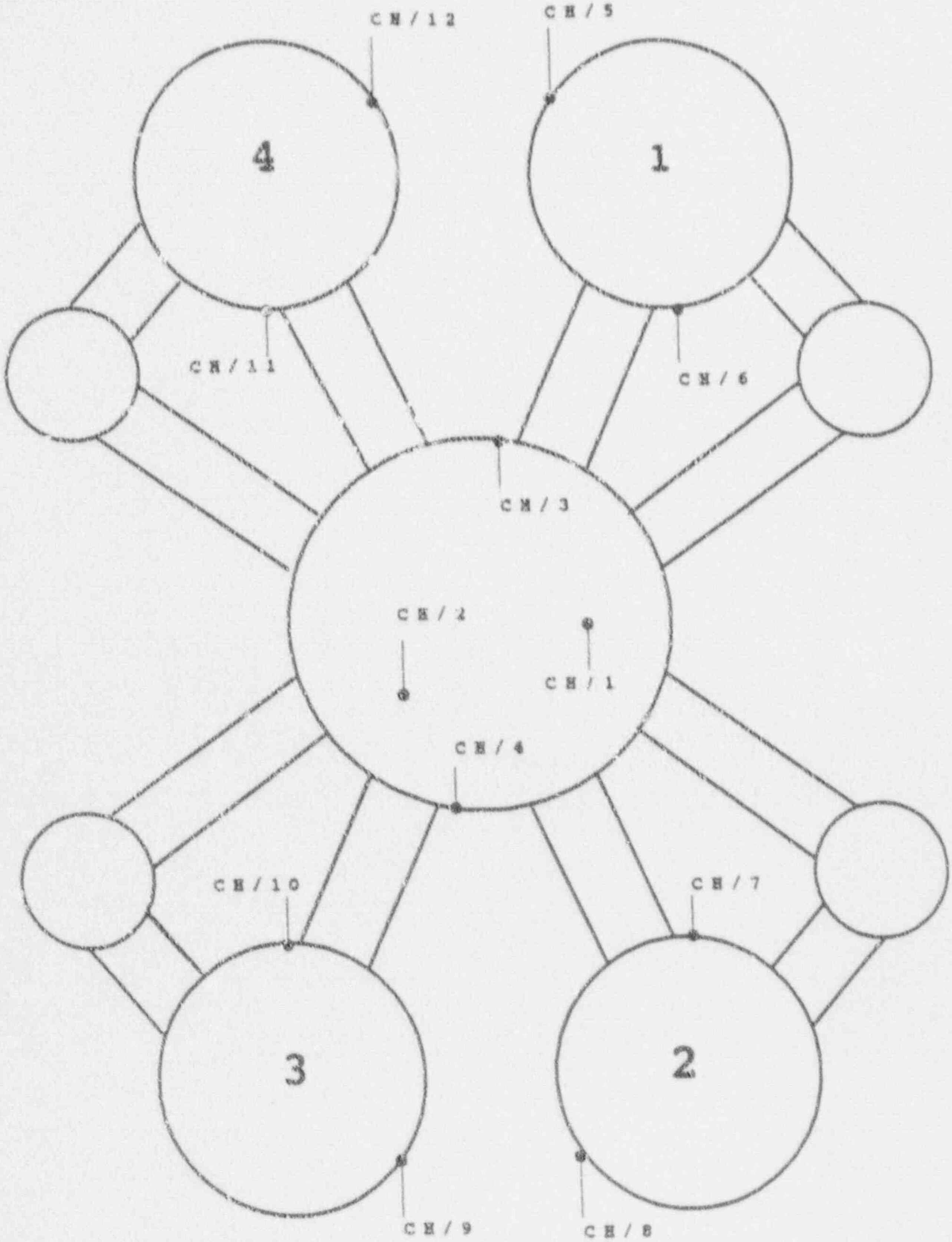
* screening criteria

III. Description of the WCNOC Investigation
(continued)

The inquiry focused on three possible mechanisms that could cause noise in the reactor coolant system:

- loose parts
- thermal-hydraulic transient
- thermal growth binding/interference

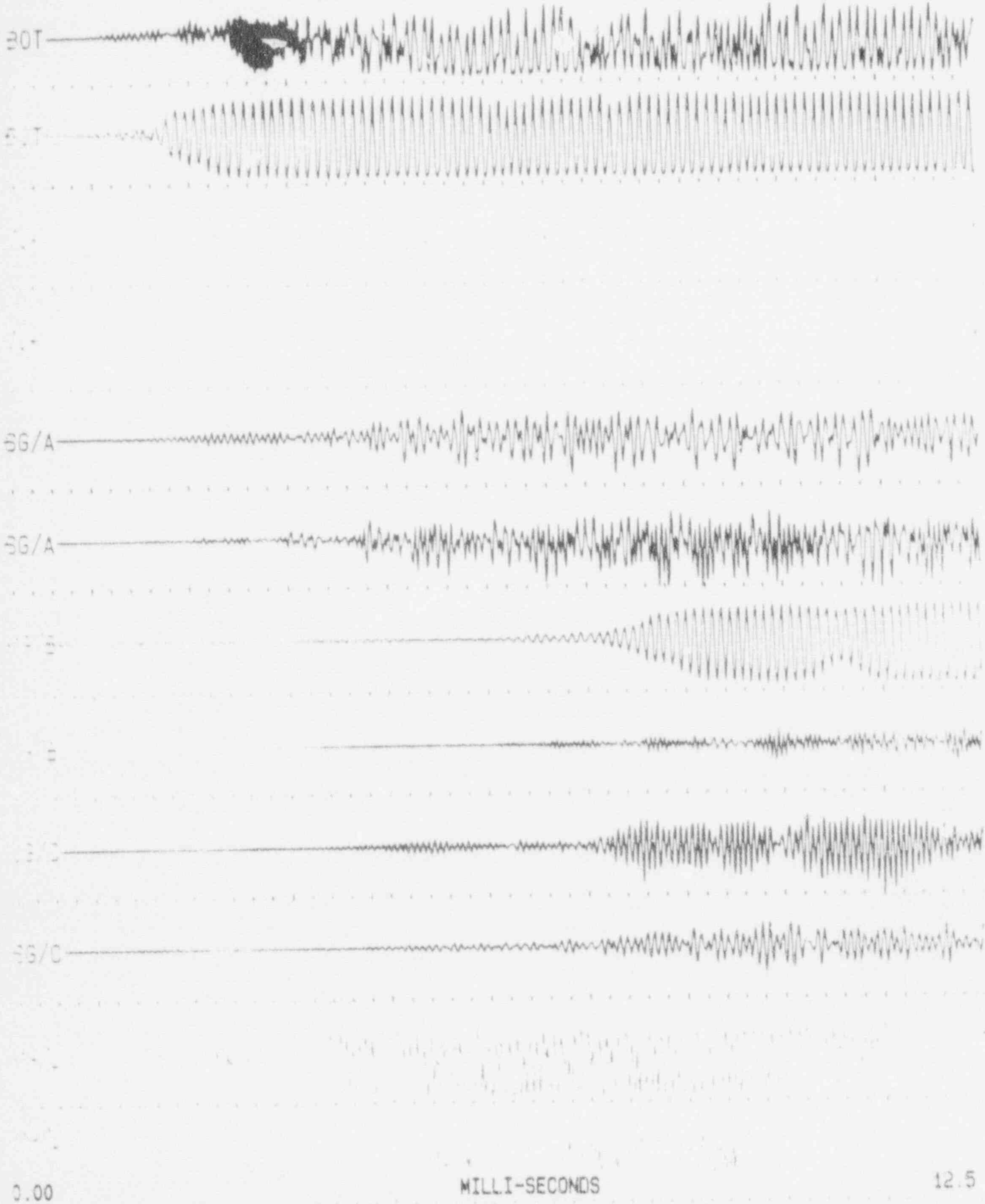
LPM SENSORS



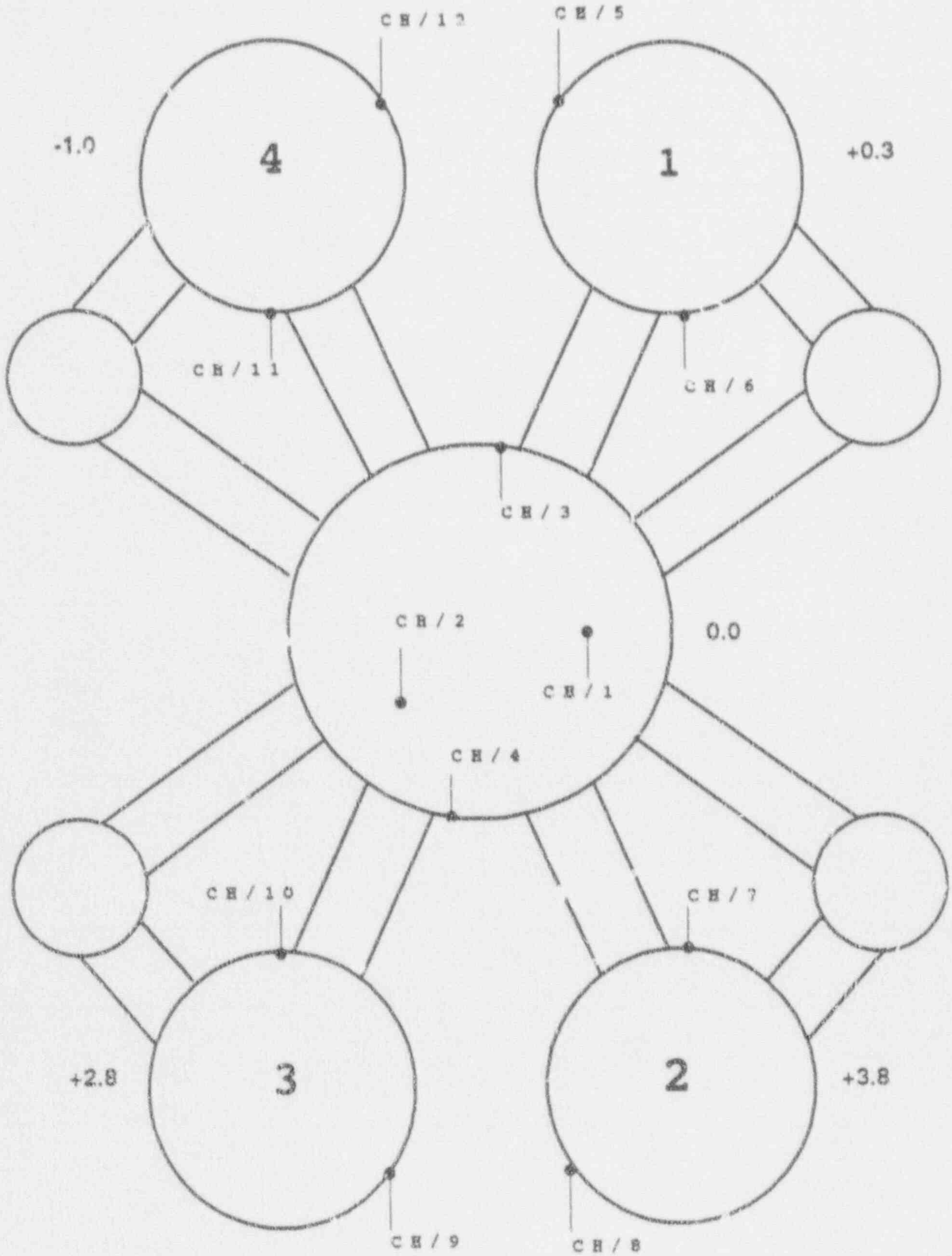
Vibration and Loose Part System

Tape Turn-On Switching Sequence

First-On Alarmed Channel	Tape Recorder Channel A	B	C	D
1	1	2	3	4
2	2	3	4	1
3	3	4	1	2
4	4	1	2	3
5	5	2	4	6
6	6	1	3	5
7	7	2	4	8
8	8	1	3	7
9	9	2	4	10
10	10	1	3	9
11	11	2	4	12
12	12	1	3	11


BOT
SG/A
SG/A
SG/C
MILLI-SECONDS
12.5

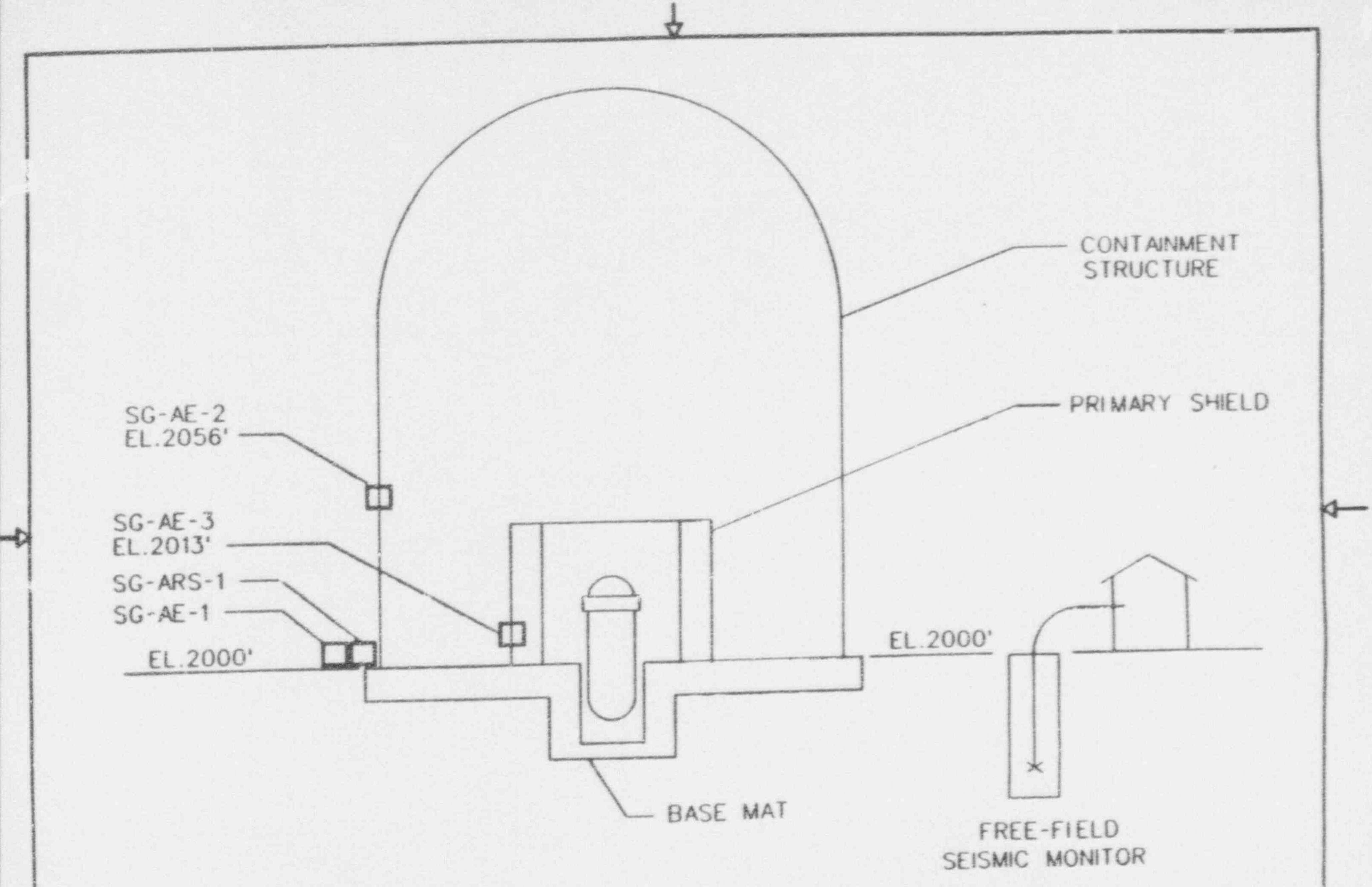
LPM SENSORS


CH/12
CH/5
-1.0
4
1
+0.3
CH/11
CH/6
CH/3
CH/2
0.0
CH/1
CH/4
CH/10
CH/7
+2.8
3
2
+3.8
CH/9
CH/8

Loose Parts Data Analysis

Conclusion

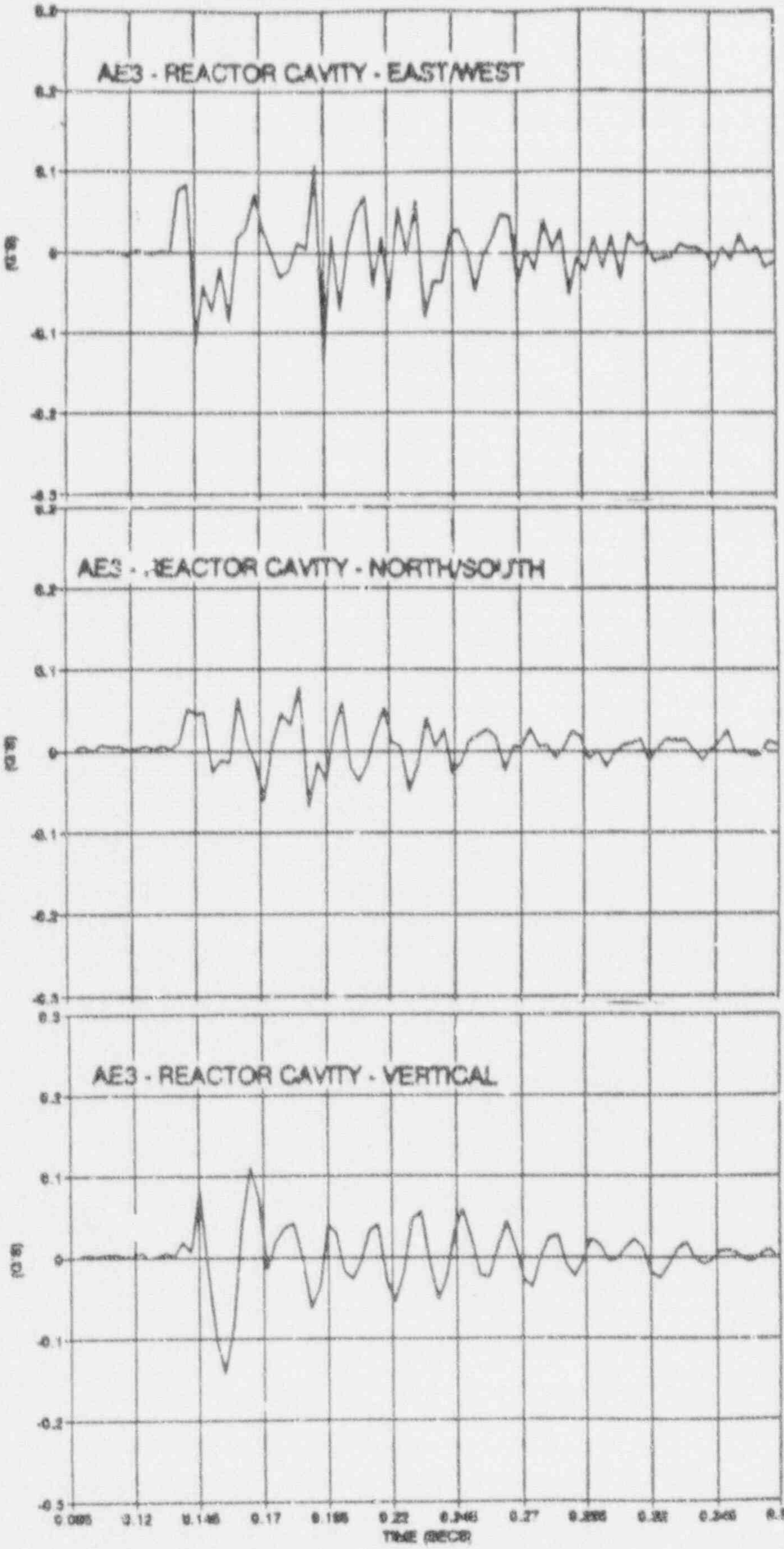
- No Loose Parts in Reactor Vessel or Steam Generators
  - Timing
  - Repetition Rate
  - Signal Character
  - Initiation Mechanism
  - Integrity Checks
- Timing Traces Suggest 3/92 Event Initiated in Reactor Coolant Loop 4



WOLF CREEK
SEISMIC MONITORING SYSTEM

DIRECTION OF INITIAL DISPLACEMENT

MARCH EVENT



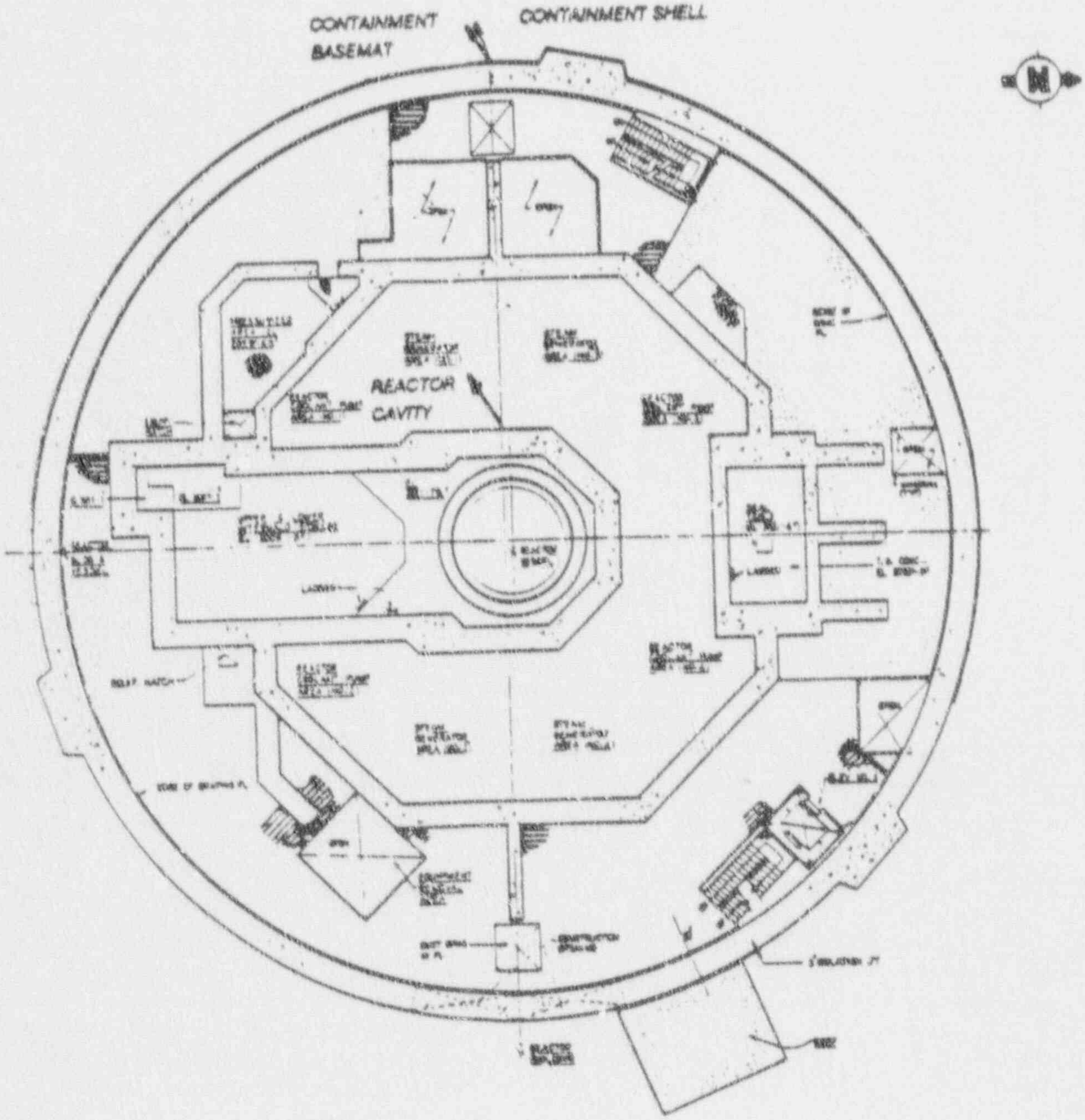
Summary of Seismic Instrumentation Response

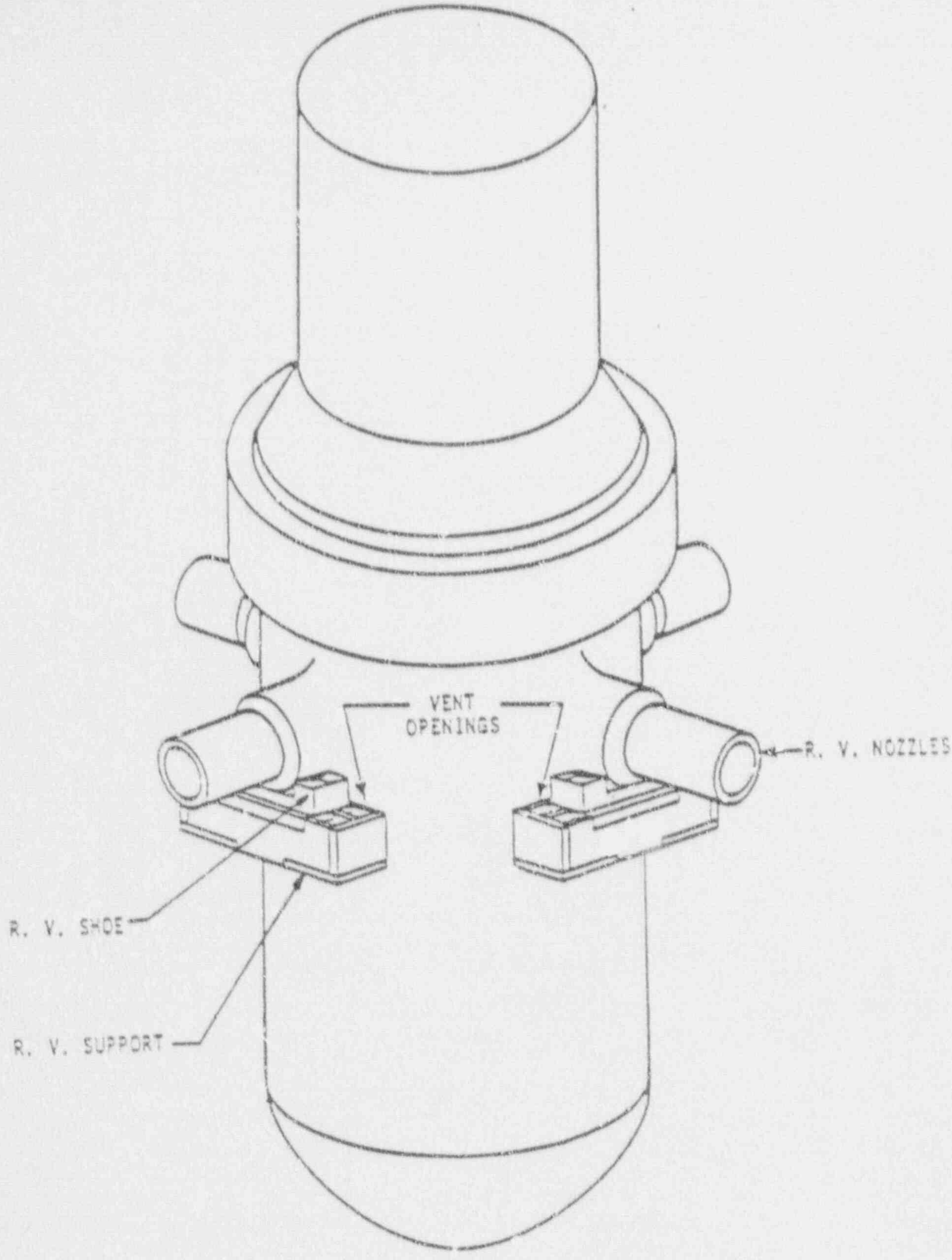
Triaxial Accelerometers:

	Reactor Cavity			Containment Basemat			Containment Shell		
	3/16	2/28	1/9	3/16	2/28	1/9	3/16	2/28	1/9
Vertical									
Peak "Gs"	.16	.16	.22	.015	.004	.005	.015	.017	.012
Est. Displ. (mils)	.4	.8	1	.2	.05	.1	.6	.4	.3
E/W									
Peak "Gs"	.135	.25	.22	.029	.026	.013	.027	.049	.02
Est. Displ. (mils)	.5	1	1	.3	.4	.3	.3	.5	.3
N/S									
Peak "Gs"	.08	.16	.11	.018	.008	.015	.011	.011	.009
Est. Displ. (mils)	.2	.9	2	.05	.04	.1	.1	.2	.03

DIRECTION OF INITIAL DISPLACEMENT

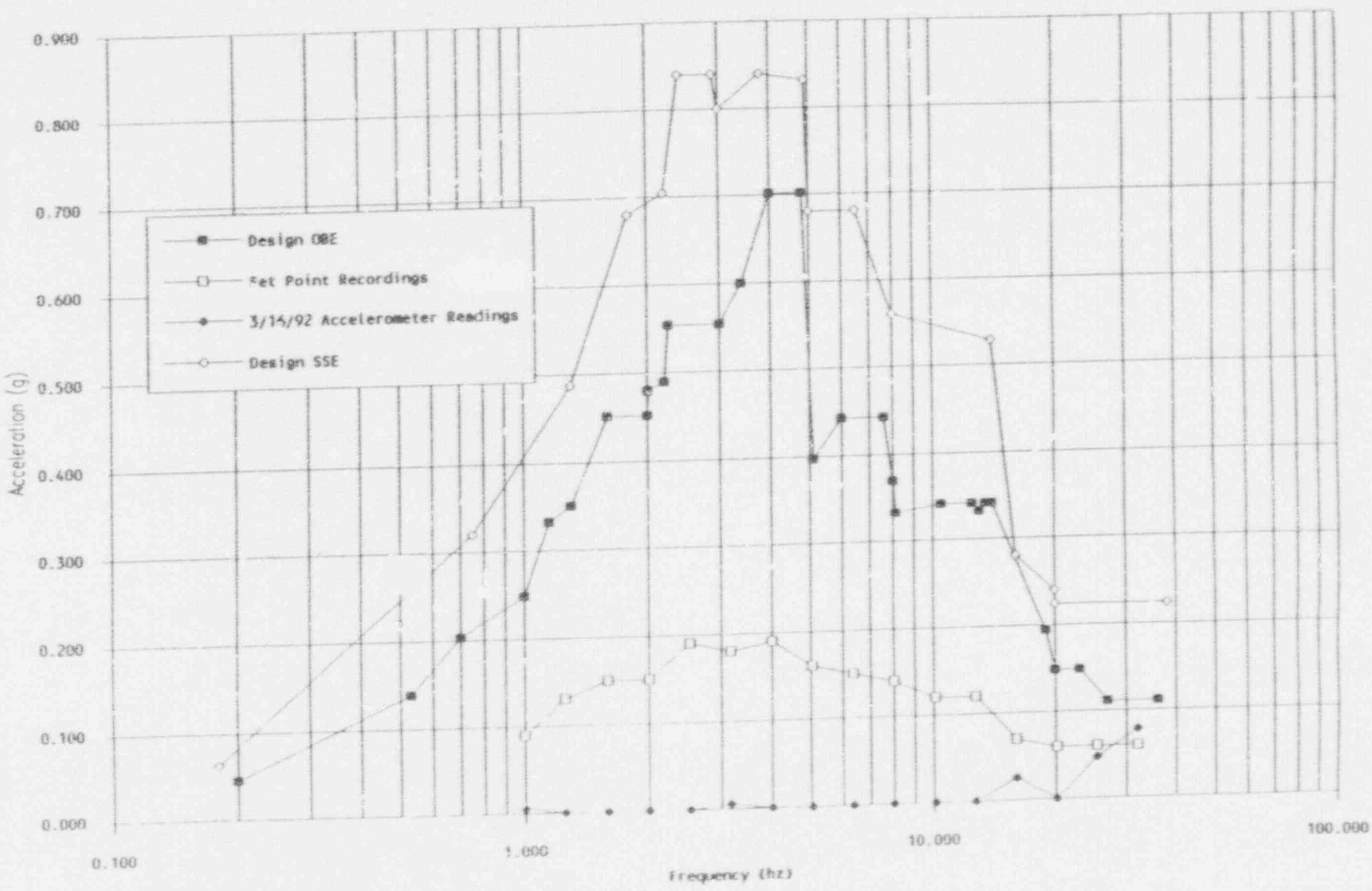
MARCH EVENT





REACTOR VESSEL SUPPORT SYSTEM

COMPARISON OF DESIGN RESPONSE SPECTRA WITH RECORDED DATA - E/W DIRECTION - EL. 2000



CONCLUSIONS

- LOADS PRIMARILY TRANSMITTED TO THE SHIELD WALL

- INITIAL MOVEMENT TOWARD WEST-SOUTHWEST

- NO STRUCTURAL DISTRESS IS INDICATED

III. Description of the WCNOC Investigation
(continued)

Thermal-hydraulic Event

The postulated event is that a pressure wave propagated through the primary system. The wave was caused by the collapse of a saturated or superheated steam bubble.

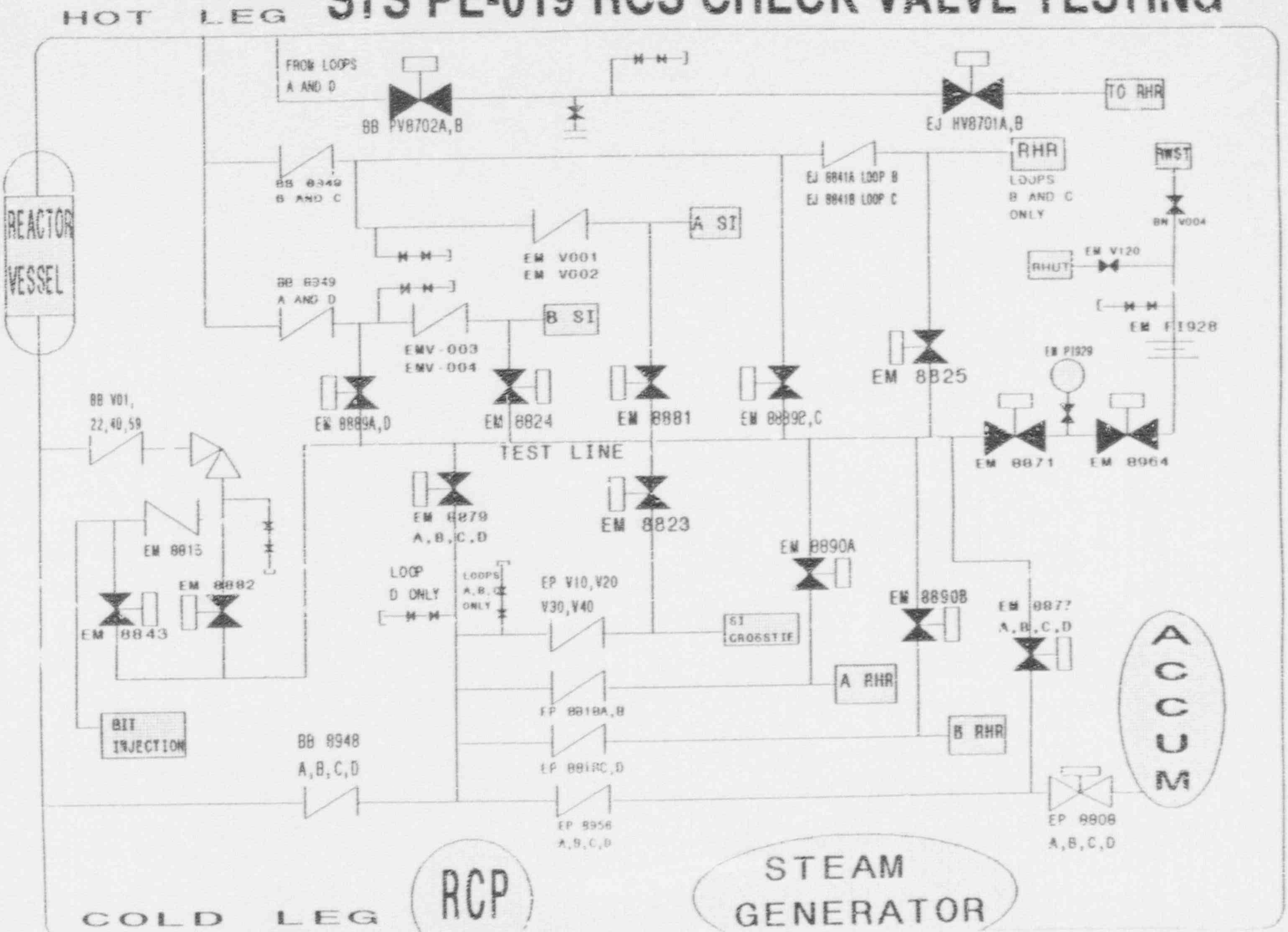
To test this mechanism

- installed additional pressure instrumentation on the safety injection system to measure conditions in the system and possible impacts on the reactor coolant system (provided data on conditions during the March 16 noise event)
- performing modelling of the saturated and superheated steam bubble situations
- tested check valves during restart (no problems noted)

Based on the data provided from the above, concluded that this was not the mechanism that caused the noise.

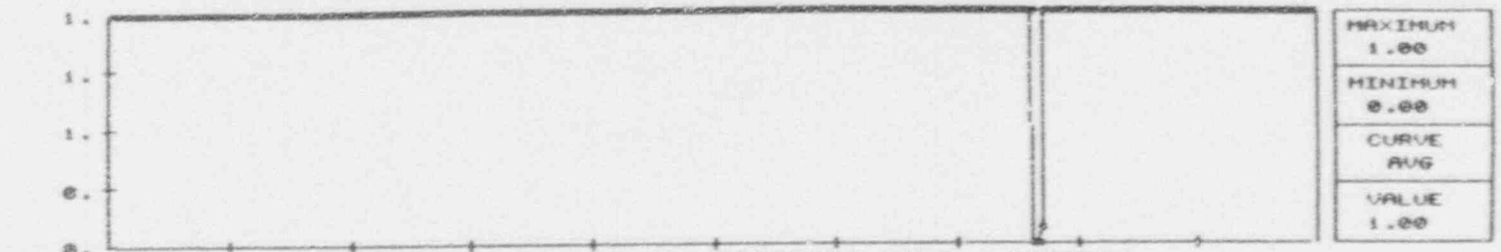
Changed check valve test program to lessen the possibility of a thermal hydraulic event when restoring the accumulators to service. The March heatup confirmed the effectiveness of these measures.

STS PE-019 RCS CHECK VALVE TESTING

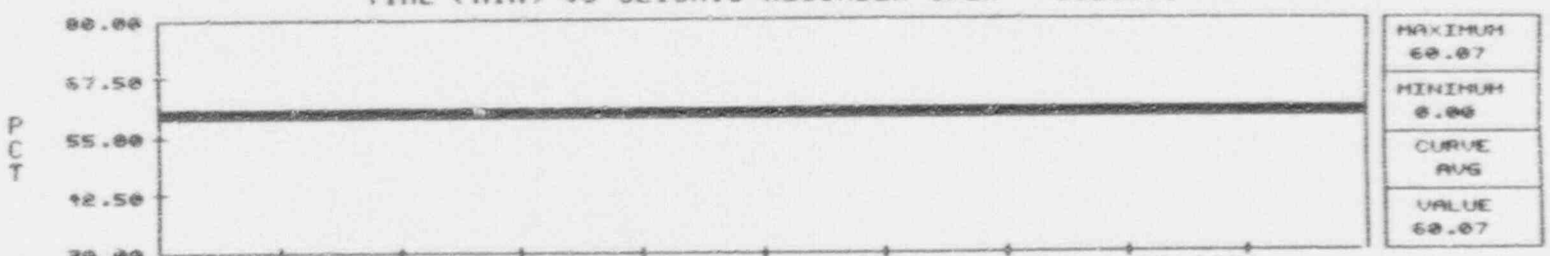


SELECT FUNC. KEY OR TURN-ON CODE TT: [redacted] 1/ 9/92 03:27:00 [redacted]

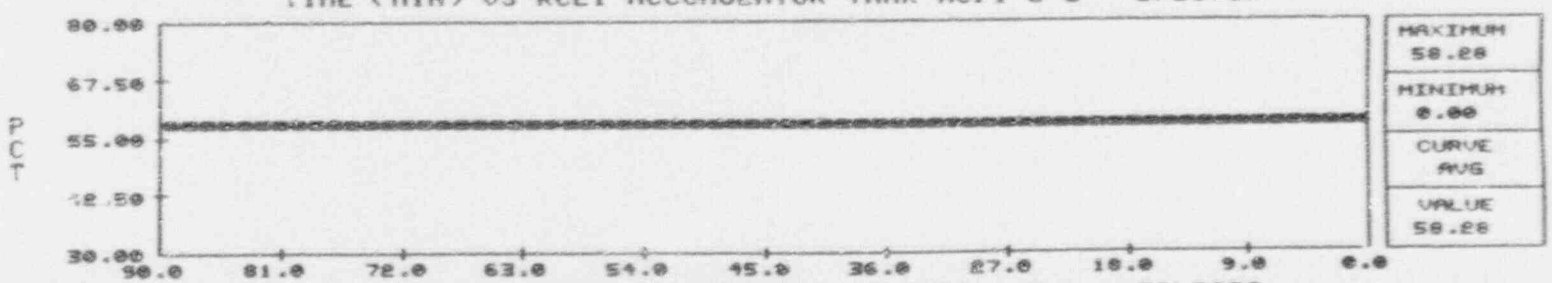
TIME TREND FOR TPHEPA



TIME (MIN) VS SEISMIC RECORDER OPER - SGD0001



TIME (MIN) VS RCL1 ACCUMULATOR TANK NO.1 B L - EPL0951



TIME (MIN) VS RCL1 ACCUMULATOR TANK NO.1 A L - EPL0950

[redacted]

F1=NEW PLOT F2=RESCALE X F3=RESCALE Y F4=QUICK PLT F5=NEXT PLOT F6=PLOT DEF.
PREV CANC W.C. TERM=TT17 CPU=A CONSOLE=PLAYBACK MODE=HOT STBY EVENT=AUTO
VIDEO COPY

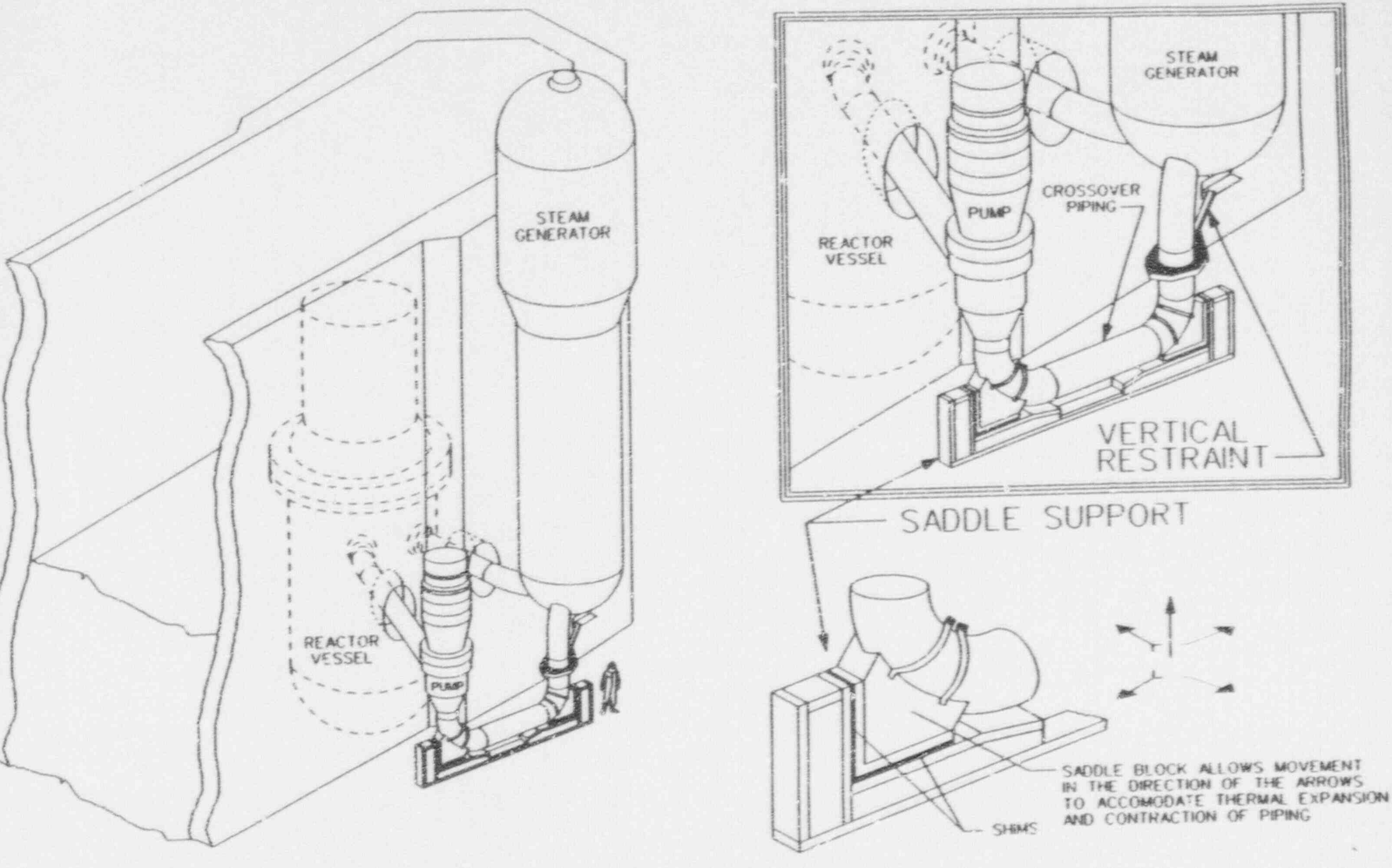
III. Description of the WCNOC Investigation
(continued)

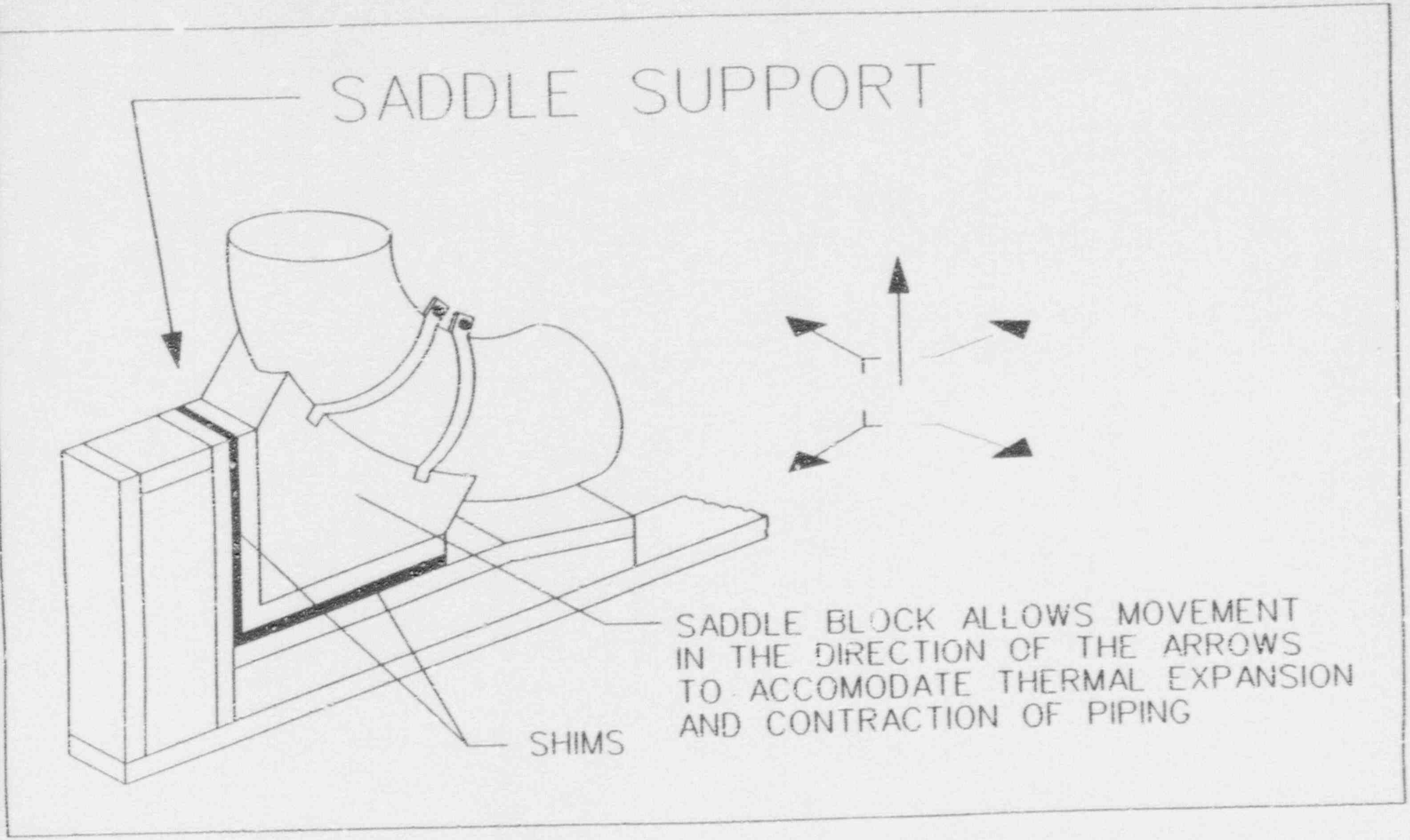
Thermal Growth

> The postulated event here is a pipe, component, or support of the primary system, as it expanded during heatup, came into contact with another component of the system or containment structure and bound up. The resulting "give" caused the noise.

The investigation included a detailed inspection of the primary system and its support structures to identify interference points, as well as indications of mis-aligned supports. Evidence of damage or contact would be the key indicator that this mechanism initiated the noise.

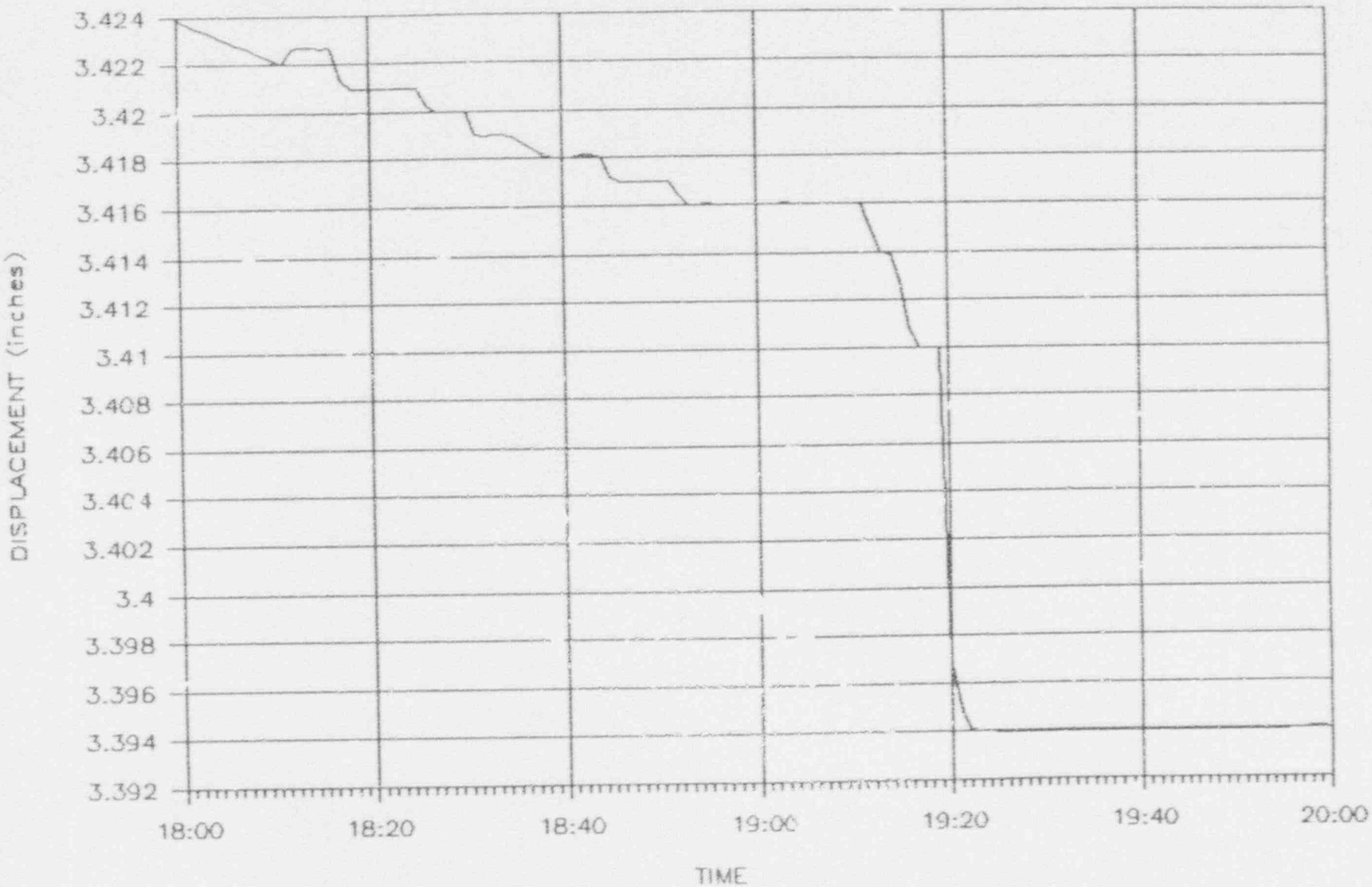
The similarities of the January '92, February '92 and March '92 events (nearly same temperature and pressure, no other activities going on) suggest that pipe growth was a very likely initiator of the noises.


STEAM GENERATOR
REACTOR VESSEL
PUMP
STEAM GENERATOR
CROSSOVER PIPING
PUMP
REACTOR VESSEL
VERTICAL RESTRAINT
SADDLE SUPPORT
SADDLE BLOCK ALLOWS MOVEMENT IN THE DIRECTION OF THE ARROWS TO ACCOMODATE THERMAL EXPANSION AND CONTRACTION OF PIPING
SHIMS


SADDLE SUPPORT
SADDLE BLOCK ALLOWS MOVEMENT IN THE DIRECTION OF THE ARROWS TO ACCOMODATE THERMAL EXPANSION AND CONTRACTION OF PIPING
SHIMS

LOOP 4

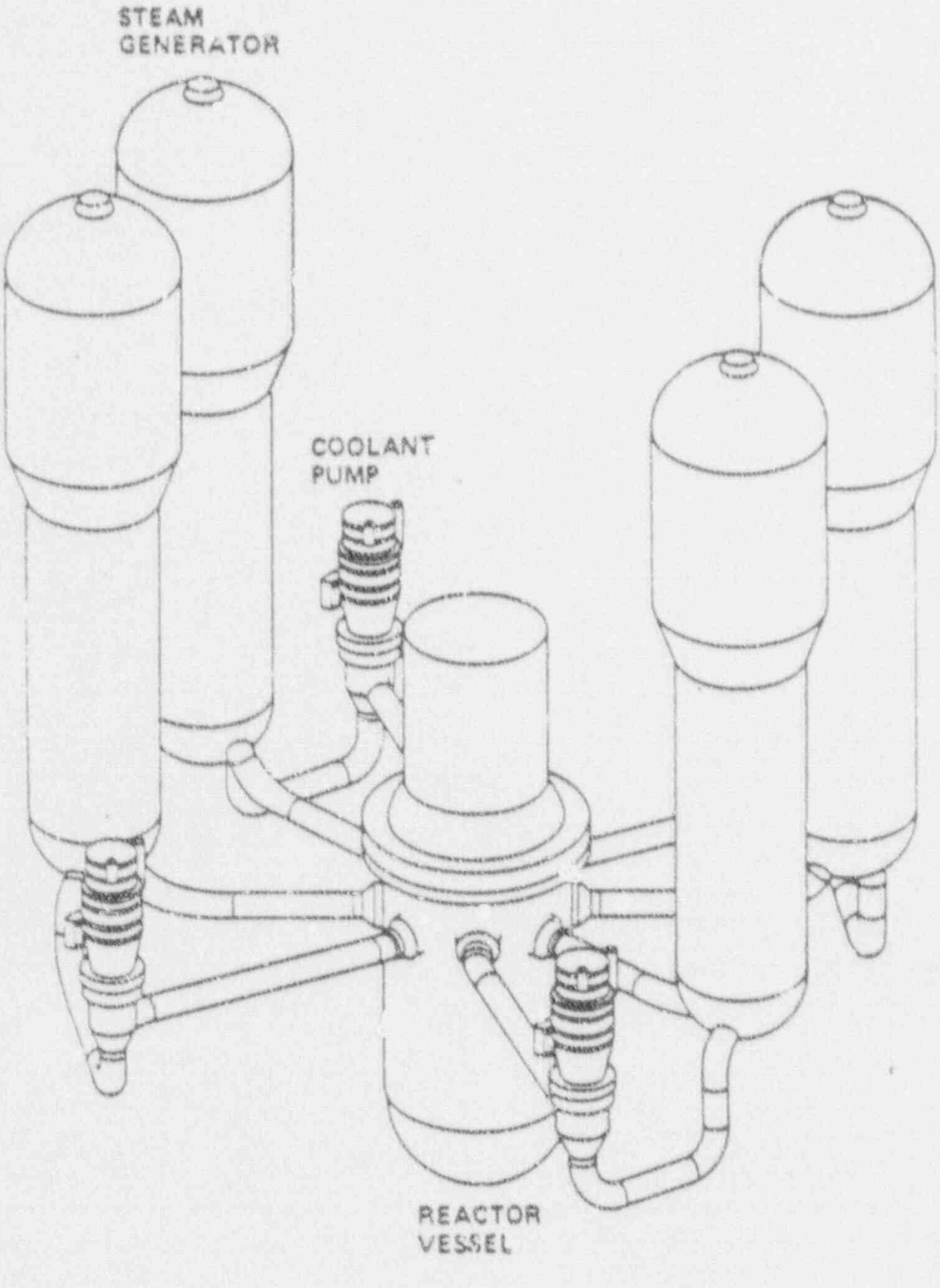
LANYARD DATA



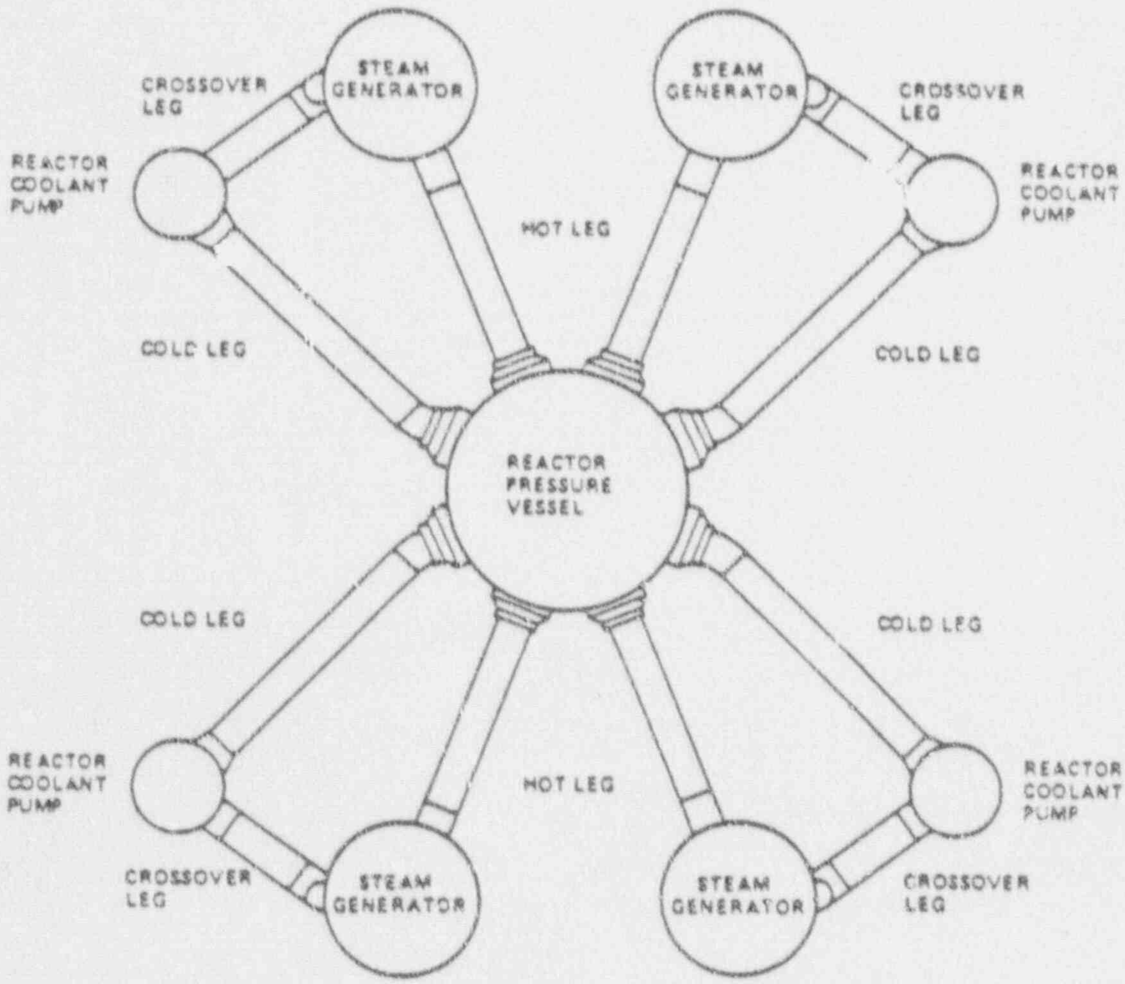
IV. The Identification and Correction of the Cause

An inspection of the shims in the crossover leg pipe whip restraint revealed evidence of hard contact.

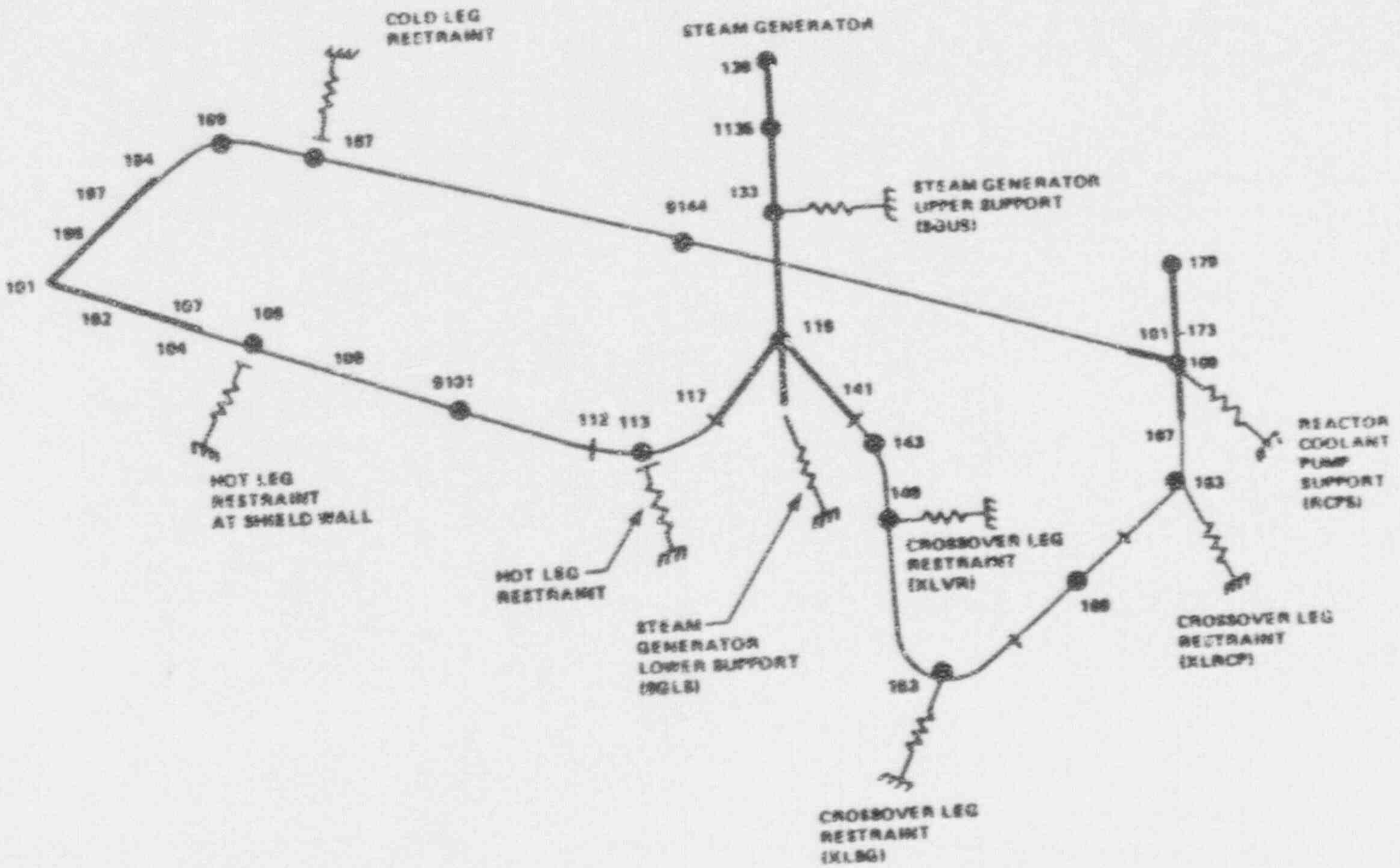
To eliminate this as a cause, the shims were removed and milled to restore the desired clearance.



SIMPLIFIED DIAGRAM OF THE NSSS



PLAN VIEW OF THE REACTOR COOLANT LOOPS



Reactor Coolant Loop Model

RCS Integrity Evaluation

- Piecewise Linear Thermal Expansion Analysis
- RCL Snapping Analysis for Potential Energy Release
- Reshimming of Crossover Leg Bumpers
- Over Temperature Transients and RCS Integrity
- Primary Equipment Support Qualification
- Fatigue Analysis
- Primary Equipment Nozzle Loads

V. Safety Impact and Plant Operability

The noise events did not cause damage to the plant or its components.

- Detailed visual inspections found no damage
- Loose parts analysis revealed no loose parts
- Calculated loadings and stresses well within safety margins

Required systems are operable

- Performed required surveillances

No damage can result from noise events of this type

- Performed bounding calculations
  - Additional seismic loading from RCS system shake
  - Maximum water hammer forces from accumulator discharge
  - Reestablish or verify support clearances

PSRC and NSRC Review Completed

Longer Term Actions

- Monitoring during Power Ascention
- Measurements/Monitoring during next heatup cycle
- Procedure Enhancements
  - Seismic Alarms
  - Check Valve Testing
- Evaluate removal of shims

VI. Summary

- Performed an exhaustive investigation assisted by outside experts
- Verified no damage occurred - verified reactor coolant system integrity
- Verified similar noise events would not cause damage
- Long term monitoring and action plan
- WCNOC took conservative and prudent measures
- Wolf Creek Generating Station is ready to resume operations