UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

AUGMENTED INSPECTION TEAM (AIT) INSPECTION

Report Nos.: 50-261/93-34

Licensee: Carolina Power and Light Company

Docket Nos.: 50-261

License Nos: DPR-23

Facility Name: H. B. ROBINSON UNIT 2

Inspection Conducted: November 20-December 6, 1993

Team Leader: T A Peebles 12/28/93
Thomas A. Peebles, Chief
Operations Branch,
Division of Reactor Safety
Date Signed

Team Members:

M. E. Ernstes, Operator Licensing Examiner
E. D. Kendrick, Nuclear Engineer
S. M. Matthews, Quality Assurance Engineer
C. R. Ogle, Resident Inspector
B. H. Rogers, Reactor Engineer
J. E. Tedrow, Senior Resident Inspector

Approved by: for Johns P. Jaudon 12/29/93
Albert F. Gibson, Director,
Division of Reactor Safety
Date Signed

EXECUTIVE SUMMARY

The objectives of the inspection were to determine the scope and the causes of the events observed during the post refueling startup of H. B. Robinson Unit 2 and to evaluate the licensee's response to these events.

H. B. Robinson Unit 2 went critical on November 12, 1993. Criticality parameters were within the expected range, and initial physics testing did not reveal any core anomalies. On November 14, 1993, with the reactor at an indicated power of 20 percent, a heat balance - done in response to management questions about diverse power indications - showed that the actual power was 30 percent.

On November 16, 1993, flux mapping indicated core peaking factor problems. These problems were confirmed by a second flux map. The licensee and the fuel supplier (Seimens Power Corporation) discovered on November 18, 1993, that six fuel assemblies had been misconstructed in that asymmetrically loaded, burnable poison was incorrectly positioned in the core. The reactor had been shut down November 17, 1993, in order to repair a steam leak in the secondary plant.

The Augmented Inspection Team was chartered on November 19, 1993, and was onsite November 20-24 and November 29-December 2, 1993. Additionally, members of the inspection team were at the Seimens' Richland, Washington facility November 29-December 3, 1993, and a public exit meeting was held December 6, 1993.

The principal findings and conclusions of the Augmented Inspection Team were:

1. Licensee oversight and assessment of the fuel constructor, refueling activities, startup preparations (including the calibration of nuclear instruments and operator training), and the conduct of operations during the startup were deficient.

2. Power range nuclear instruments had been mis-calibrated because of an inadequate understanding of the core geometry, and the operators did not diagnose the incorrectly reading power range instruments, although there were sufficient indications available in the control room. Specifically, lessons learned from an event at another plant in which power range nuclear instruments were not reading correctly were not effectively utilized to prevent this occurrence.

3. The plant operated with fuel having mis-positioned burnable poison. This did not result in fuel damage, but damage could have occurred if the plant had operated above 30 percent.

4. The six misconstructed fuel assemblies were the result of inadequate fabrication controls and oversight by the fuel supplier.

5. The licensee's post event review and evaluation were adequate.

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Background

During the restart of H. B. Robinson Unit 2, following the cycle 15 refueling outage, which refueled the unit with the Cycle 16 core, the NRC was aware of the following sequence of events:

DATE	TIME	EVENT
11/12/93	12:14 am	Commenced reactor startup.
11/12/93	6:19 am	Reactor critical. Estimated Critical Position met.
11/14/93	7:00 am	Reactor at the point of adding heat.
11/14/93	8:00 am	Intermediate Range nuclear instrumentation NI-36 bypassed.
11/14/93	9:22 am	Operations realized that there was a reactor power mismatch: 30 percent actual; 20 percent indicated.
11/14/93	9:00 pm	Loose Parts Monitoring System discovered deenergized.
11/14/93	Evening	Site team formed to review startup.
11/16/93	4:00 am	Flux map indicates abnormal peaking factors. (Core design problem.)
11/16/93	6:15 pm	Leaking weld identified (at FW-13B).
11/16/93	8:40 pm	Second flux map confirms peaking factors.
11/17/93	1:13 am	Started unit shutdown to repair FW-13B.
11/17/93	Morning	Licensee management review team on-site to investigate.
11/18/93	Afternoon	Licensee and fuel vendor decide that core misloading is the cause of the core peaking factor problem.

1.2 AIT Formation

On November 19, 1993, senior NRC managers concluded that events surrounding this startup warranted further independent evaluation; an Augmented Inspection Team was formed, and a Confirmatory Action Letter was issued by Region II. A detailed charter was developed to guide the team. In addition to the above events, loose parts had been identified during fuel handling activities in the spent fuel pit. Inspection of this item was included in the charter (the AIT

Charter is Appendix B). The team began its inspection on site on November 20, 1993.

2.0 EVENT DESCRIPTION

The following is the detailed sequence of events associated with the November 12, 1993, startup until Hot Shutdown was reached on November 17, 1993. The team found no major disagreements from the licensee's sequence of events. A detailed narrative begins at paragraph 3.0.

DATE	TIME	EVENT
11/12/93	12:14 am	Shut reactor trip breakers, commenced reactor startup per Procedure GP-003.
11/12/93	12:20 am	Commenced Procedure EST-050, Refueling Startup Procedure. (Control Rod withdrawal then dilute to criticality.)
11/12/93	6:08 am	Deenergized Source Range Nuclear Instruments, and NI-32 hung up at 45 cps.
11/12/93	6:18 am	Start Up Rate meter pegged high. Stopped Procedure EST-050. Power stabilized at 10^{-8} amps in the Intermediate Range.
11/12/93	6:19 am	Reactor critical.
11/12/93	11:49 am	Cleaning of Start Up Rate meter selector switch complete.
11/12/93	1:00 pm	Recommenced Procedure EST-050
11/12/93	2:00 pm	Commenced low power physics testing per Procedure EST-050.
11/13/93	11:40 am	Completed low power physics testing.
11/13/93	1:28 pm	Reactor at one percent.
11/13/93	2:07 pm	NI-44 returned to service following Procedure EST-50.
11/14/93	12:52 am	NI-35 Out Of Service for setpoint changes.
11/14/93	3:12 am	NI-35 returned to service.

11/14/93	3:14 am	NI-36 Out Of Service for setpoint changes.
11/14/93	3:55 am	NI-36 returned to service.
11/14/93	6:39 am	Latched main turbine.
		<u>Operator Distractions</u>
		Left main turbine stop valve bypass won't open due to isolated instrument air valve, IA-3221, shut. Unable to bypass around and equalize across left stop valve.
		Watchstation turnover.
		Turbine rolling approximately 200 rpm due to leakage past the governor valves. Operators concerned over Procedure GP-005 precaution to minimize time turbine below 200 rpm.
		Turbine vibration alarms occurred.
11/14/93	7:47 am	Turbine valve trip test. Number two intercept reheat valve does not indicate closed.
11/14/93	7:51 am	Turbine relatched.
11/14/93	7:54 am	Turbine at 1800 rpm.
11/14/93	8:08 am	Unit on line. Power escalation commenced.
		<u>Operator Distractions</u> Sluggish voltage regulator response.
		No Mega Volt Amperes Reactive indication on gauge board.
		Load dispatcher reports telemetry failure.
		All four turbine governor valves indicate shut.
11/14/93	8:15 am	Intermediate Range NI-36 is bypassed at Nuclear Instrumentation cabinet to preclude reactor trip.

Operator Distractions

Low level in "A" Steam Generator.

No feed flow indication on A & C Steam Generators.

Steam flow greater than feed flow alarms.

Feed Water Heater Alarms.

Swap of electro-hydraulic oil pumps due to A pump not unloading.

Main steam reheater vent to condenser valve (FCV-1334) indication problem. Balance of plant operator required to hold valve switch on gauge board to open.

System engineer reports turbine vibrations recorded in control room twice field reading.

Condensate header low pressure alarm due to condensate pump recirculation valve FCV-1446 hung open.

System engineer identifies one of four generator H_2 coolers isolated.

11/14/93	8:42 am	Main Feedwater regulating valves in auto. Power escalation stopped to stabilize reactor power.
11/14/93	8:57 am	Reactor power stabilized at 20 percent as indicated by Power Range Nuclear Instrumentations
11/14/93	9:02 am	Electrical operator questions indicated Nuclear Instrumentation power based on turbine first stage pressure equates to approximately 26 percent power.
11/14/93	9:22 am	Engineering Technical Services manager questions difference between reactor power indicated by Power Range Nuclear Instrumentations and net Mega Watts electric. Operators estimate 30 percent reactor power based on loop delta Ts.

11/14/93	9:40 am	Generator volt-ampere (mega volt amperes reactive) knife switch found open. Closed switch and restored mega volt amperes reactive indication.
11/14/93	10:26 am	Initial calorimetric data per Procedure OST-10 indicates reactor power is 30.26 percent.
11/14/93	10:44 am	Operations Manager and Engineering Technical Support Manager notified of initial calorimetric results per Procedure OST-10 Reactor engineer support requested.
11/14/93	10:47 am	Decreased reactor power to less than 30 percent.
11/14/93	11:01 am	Procedure OST-10 complete. Calculated reactor power is 30.26 percent.
11/14/93	12:20 pm	Regulatory affairs notified of potential TS 3.10.7.1 violation for exceeding three percent/hr ramp rate immediately following refueling.
11/14/93	2:00 pm	Procedure EST-53 indicates actual reactor power is 30.03 percent.
11/14/93	2:57-3:12 pm	Instrumentation and controls personnel recalibrated Power Range Nuclear Instrumentations.
11/14/93	9:00 pm	Loose parts monitor discovered disabled. When placed in service, alarm received on primary side of Steam Generator C.
11/14/93	11:30 pm	Noise from potential loose part on Steam Generator C subsided.
11/16/93	4:00 am	Operations notified that flux map results identified peaking factors that require additional analysis.
11/16/93	6:10 am	NI-35 out of service to reset high flux trip and rod stop.
11/16/93	5:33 pm	NI-35 high flux trip and rod stop reset. No retest due to procedural problems and plant conditions.

11/16/93	8:10 pm	Feedwater leak identified on main feedwater pump "A" discharge. Second flux map indicates core flux irregularities.
11/17/93	12:35 am	Feedwater leak traced to weld crack for feedwater drain valve, (FW-13B). Crack growing.
11/17/93	1:13 am	Unit shutdown commenced at one percent/min per Procedure GP-006.
11/17/93	1:35 am	Station output breakers opened.
11/17/93	4:14 am	Reactor shutdown.
11/17/93	8:39 am	NI-35 failed Procedure OST-001.
11/18/93	11:16 am	Completed Shutdown Procedure GP-006.

3.0 MISCALIBRATED NUCLEAR INSTRUMENTS

3.1 Review of the root cause of the miscalibrated nuclear instruments identified during startup.

During the plant startup on November 14, 1993, licensee personnel discovered that power range nuclear instrumentation indicated power readings were approximately ten percent lower than actual reactor power. The licensee attributed the cause for this discrepancy to be the effect that the new reactor core had on the power range nuclear instrument indication and an improper understanding of core geometry considerations. During the cycle 15 refueling outage, the licensee installed a very low leakage core. The new core load not only reduced the neutron flux present at the reactor vessel, but also reduced the neutron flux which reached the nuclear instrument detectors located on the outside periphery of the vessel. To account for this change in the neutron flux, the intermediate range and power range nuclear instruments were recalibrated to adjust the previous cycle instrument currents to predicted new cycle instrument currents.

The team reviewed the licensee's startup testing which was performed following the refueling outage and the associated nuclear instrumentation work packages which documented the calibration of the nuclear instruments with particular emphasis on the intermediate range and power range detectors. In addition, the licensee's investigation into this event was reviewed.

Procedure FMP-002, "Nuclear Instrumentations Post Refueling Adjustment Determination," was implemented by the licensee to quantify the projected impact on the nuclear instrumentation by the new core loading. The team verified the licensee's calculations in this procedure for the new power range 100 percent currents and the intermediate range rod block and high level trip setpoints. The team reviewed Work Request WR 93BMZ001 and verified that the 100 percent currents were implemented into the power range channels.

Westinghouse issued a letter to the licensee on March 16, 1988, providing guidance on nuclear instrumentation concerns when implementing a core design change which reduces the neutron leakage from the core. This letter was initiated because of core design changes made at the licensee's Harris facility. The letter recommended a correction factor which was calculated from the average relative power of four fuel bundles, which comprised the complete outer diagonal nearest the power detector. This guidance was not incorporated at plant Robinson even though corporate licensee personnel were aware of the Westinghouse letter which was implemented at the Harris facility. Instead, licensee personnel developed the calculations contained in Procedure FMP-002 based on previous cycle performance data which agreed more closely with historical core data. These calculation utilized the two nearest outer diagonal fuel assemblies and a third inner assembly (second diagonal) for relative power comparisons.

The licensee determined that their method was in error following communication with the fuel vendor, who stated that the outer fuel assemblies contributed approximately 16 percent each to the flux indicated on the detector while the inner assembly contributed only 2-3 percent. The incorrect methodology was not detected during previous startups because previous core loadings fortuitously had fuel assemblies with similar relative power loaded in the inner position and on the periphery of the outer diagonal, thereby canceling any mathematical averaging errors. The new, low leakage core on the other hand, specified that a much higher average relative power assembly be loaded in the inner position which differed substantially from the outer diagonal assemblies. When the averaging calculations were performed for the new core load, the error in methodology caused a discrepancy between predictions of approximately 90 percent. Based on the information from the licensee's vendors, the team concluded that the root cause for the nuclear instrument miscalibration was due to the incorrect methodology used in calculating the power range currents. This was confirmed by licensee personnel who calculated the predicted power range indication using the correct methodology. These calculations indicated that when actual power was 30 percent the correctly calibrated power range instruments would have indicated approximately 38 percent, which would have been conservative and acceptable.

Also during this outage, both of the source range and intermediate range nuclear instrument detectors were replaced due to aging. The methodology used in the intermediate range calculations agreed between Westinghouse and the licensee and therefore the calculations for intermediate range rod stop and high level trip setpoints were unaffected.

As part of the startup test program, Procedure EST-050, Refueling Startup Procedure, was used to establish the high power reactor protection system trip setpoint at 45 percent. The TS limit for this setpoint is 109 percent power. The action to reduce this setpoint was taken in response to a similar nuclear instrument miscalibration which occurred in December 1998 at the licensee's Harris facility which is similar in design to the Robinson plant. The team reviewed work request WR 93HUK001 completed on November 9, 1993, which implemented the 45 percent trip setpoints. The team considered this action by the licensee to be very beneficial which would have limited a potential power excursion. The team calculated that a reactor trip would have occurred once actual power reached approximately 67 percent, which is far below the technical specification limit. The team considered the conservative action to reduce the high power trip setpoint after refueling outages to be a program strength which helped reduce the potential consequences of this event.

3.1.1 AIT Findings and Conclusions

- The team determined that improper methodology for predicting Power Range currents was used by Robinson. Inadequate corporate/site oversight and communications contributed to this use of improper methodology. The licensee corporate fuels staff was aware of the Westinghouse recommended method for predicting currents, its basis, and its implementation from prior experience at the Harris plant.

- As a safety precaution, the setting of the Power Range High Power Flux Trip had been set at 45 percent vice 109 percent prior to startup, this would have limited any power increase to less than an allowed value. However, the core flux anomalies coupled with the power range nuclear instrument misalignment would have resulted in high neutron flux in localized areas of the core if power had reached the indicated 45 percent.

3.2 Assessment of the adequacy of station nuclear instrumentation calibration and refueling procedures.

The licensee's nuclear instrument calibration procedures were reviewed as well as work packages which were performed on the nuclear instruments during the plant startup. Both of the source range and intermediate range nuclear instrument detectors were replaced. The team discussed the procedure guidance with licensee personnel and compared the procedure scope with control wiring diagrams and the system technical manual to determine completeness. The following procedures were reviewed:

- LP-702 Nuclear Instrument System Source Range
- LP-703 Nuclear Instrumentations Pulse Amplifier NM-101, Attenuation, Discrimination and High Voltage Power Supply NQ101
- LP-704 Nuclear Instrument System Intermediate Range Channels NI-35 & NI-36
- LP-705 Nuclear Instrumentations Power Range Channel NI-41, NI-42, NI-43, and NI-44
- PIC-107 Power Level Indication at the Power Range
- PIC-109 Nuclear Instrument System Over Power Trip High Range Adjustment for the Power Range Flux Detectors
- PIC-110 Nuclear Instrumentations Intermediate Range (NI-35 & NI-36) Compensating Voltage Adjustment and Loss of Compensating Voltage Alarm Adjustment

The team found the guidance provided in the licensee's calibration procedures to be adequate and closely agreed with the system technical manual. However, the team found a few deficiencies in the data sheets provided in Procedures LP-704, LP-705, and PIC-109. The data sheets were considered by the team to be confusing since procedure sections were not specifically identified for data recording. Further, the team noticed that acceptance criteria was not included on the data sheets. This information was only included in the body of the procedure. Although this practice did not prevent successful completion of the procedure, it hampered supervisory review of the completed package. Licensee personnel had already identified this matter and appropriate procedure changes were planned to upgrade the data sheets.

From a review of the work packages, the team identified implementation problems. The intermediate range nuclear instruments were not calibrated with the new rod stop and high trip setpoints calculated by Procedure FMP-002 until after the reactor was critical and at the point of adding heat on November 14, 1993. This situation was contrary to the requirements of Procedure EST-050, step 3.10, which documented that the Nuclear Instrumentation adjustments per Procedure FMP-002, had been completed prior to taking the core critical. The team discussed with the responsible individual, why this requirement was initialed as completed without the appropriate adjustments being completed. The person responsible indicated that due to miscommunication with the maintenance technicians, he believed that the adjustments had been completed. The licensee's investigation had also identified this deficiency. The team reviewed the safety significance of this situation. The intermediate range rod stop and high level trip setpoints were not required by the licensee's technical specifications. Since the setpoints which were present at the time of reactor criticality were set at the old cycle values and were lower than the new predicted currents, startup with the old setpoints was considered to be conservative by the team.

In addition, the work packages (WR 93-AJTB1, WR 93-AJBG2) associated with the replacement of the intermediate range NI-35 and NI-36 detectors, were reviewed by the team. Typically the licensee replaces the source range detectors every outage due to aging effects, and the intermediate range detectors are likewise replaced at the same time due to location. No discrepancies were identified.

3.2.1 AIT Findings and Conclusions

- Written procedures were generally considered to be adequate. Deficiencies were noted in some data sheets; for example, acceptance criteria and tolerance bands were not specified, and procedure sections were not specifically identified. The licensee had already identified these issues, and the procedures were included in an upgrade program but had not been completed prior to startup.

- Implementation problems were noted in establishing the Intermediate Range Nuclear Instruments' high level trip and rod stop setpoints prior to criticality - they were not done until the point-of-adding-heat. Also, source range NI-32 channel was not recalibrated following detector replacement even though the procedure and technical manual recommends that this should be done. Procedural and work controls were lacking; however, the old setpoints were found to be conservative by the team.

3.3 Operator Performance relative to Nuclear Instrumentation miscalibration problem.

3.3.1 Operator Interviews

At 12:14 a.m. on November 12, 1993, following the completion of Refueling Outage 15, the licensee commenced a reactor startup. The startup and subsequent power escalation were performed in accordance with three procedures: General Procedure, GP-003, "Normal Plant Startup From Hot Shutdown to Critical;" Refueling Startup Procedure, EST-050; and General Procedure, GP-005, "Power Operation." Procedure GP-003 established the initial conditions for the startup. Procedural control was transferred to EST-050 for initial criticality and zero power physics testing. The escalation of power into and through the power range was accomplished with Procedure GP-005. This sequence and other key events of the startup are documented in Paragraph 2.0. At 9:22 am on November 14, 1993, with power stabilized at 20 percent on the nuclear instruments, the Manager of Engineering Technical Support questioned the apparent mismatch between power range Nuclear Instrumentations and net Mega Watts electric. Estimates of reactor power by the operators from loop delta Ts, indicated that power was close to 30 percent. A subsequent calorimetric calculation confirmed this estimate and the power range Nuclear Instrumentations were set to thirty percent. The increase in power to 30 percent caused a violation of technical specifications in that the 3 percent per hour rate of power rise limitation between 20 percent and 100 percent of reactor power specified in Technical Specification 3.10.7,was exceeded. The actual rate of power increase was approximately 10 percent in a 15 minute period. A flux map performed at 30 percent power indicated flux tilt and anomalous power levels. The crew maintained power at 30 percent while efforts were made to resolve the flux anomalies. A second flux map provided similar results. Following the discovery of a secondary side steam leak, a reactor shutdown was commenced on November 17, 1993.

The team attributed the mismatch between the actual power and the Power Range Nuclear Instrumentation readings to be one result of an inadequate calibration procedure. This conclusion and its basis are discussed in Paragraph 3.1.

The team reviewed the startup to assess operator performance relative to the nuclear instrument miscalibration problem. This review consisted of interviews of control room operators, as well as reviews of instrument traces, plant computer printouts, the completed startup procedures and the shift supervisor and reactor operator logs.

Each watchstander interviewed, cited prevention of a plant trip as his major, if not primary function. None of those interviewed verbally attached a similar significance to monitoring instrumentation for failure or inaccuracies. This focus on preventing a trip may have resulted in key individuals concentrating on a limited range of plant parameters resulting in ineffective oversight by members of the watch section. This reduced the potential for earlier identification of the power mismatch. Site management was not aware of this focus and had not provided adequate direction to cause the shift to be also be observant of the overall plant conditions.

The shift supervisor expressed concern prior to the watch over an Intermediate Range Nuclear Instrumentation reactor trip. He experienced one during a startup at Robinson in 1988. Adding to this concern, was an E-mail message from a reactor engineer sent the previous evening, warning of potential problems in the response of the Intermediate Range Nuclear Instrumentation during the power ascension. The memo addressed Intermediate Range Nuclear Instrumentation detector setpoint adjustments and the potential for achieving the intermediate range rod stop (20 percent current equivalent on the Intermediate Range Nuclear Instrumentation) prior to satisfying the P-10 interlock (10 percent on the Power Range Nuclear Instrumentations). The memo did not mention the adjustments to the Power Range Nuclear Instrumentations performed during the outage. In essence, the memo led the operators to believe that the Power Range Nuclear Instrumentations would be a more reliable indicator of reactor power than the Intermediate Range Nuclear Instrumentation. Furthermore, the discussion in the memo on satisfying the intermediate range rod stop at 20 percent prior to satisfying the P-10 interlock at 10 percent, reduced the potential for the operators to question an apparent 10 percent mismatch between the Intermediate Range Nuclear Instrumentation and Power Range Nuclear Instrumentations. The memo reinforced the crew's concern for a trip on the intermediate range high flux prior to satisfying the P-10 interlock. For these reasons, a shift supervisor, supplementing the crew, was assigned to monitor the Intermediate Range Nuclear Instrumentation to ensure that it did not exceed the trip setpoint prior to bypassing the trip functions. This assignment prevented this individual from maintaining an overview of the plant during a portion of the power ascension.

Several watchstanders cited difficulties in control of steam generator level and Tave as significant distractions during the startup. Review of computer printouts indicated that severe oscillations occurred in the steam generator levels until automatic level control was established. These difficulties were in part complicated by the lack of feed flow indication at low power levels on two of the steam generators. One reactor operator was dedicated to control of feed flow and the steam generator water levels. The SRO was also involved with the steam generator water level control as this was recognized as having a high potential of causing a reactor trip.

The team was unable to determine categorically if these difficulties in plant control were more severe than in prior startups or severe enough to mask the power mismatch. However, the team noted that even if the entire efforts of the three control board operators were directed at plant control problems, three shift supervisors and a shift technical advisor were still present to perform oversight and overview of the startup.

Throughout the interviews, watchstanders identified distractions as detracting from their efforts during the startup. The major distractions that occurred at key points in the startup are included in the timeline discussed in Paragraph 2.0. While it is obvious that any distraction would impact operator performance during the startup, the team was unable to assess the severity of this impact. The team did note that the operator's self-assessment of the impact of these distractions covered a broad spectrum from severe to minimal impact.

A thorough, pre-evolution brief was not conducted coincident with watch relief immediately prior to the power escalation. The team identified that the crew did not review the precautions in Procedure GP-005 in detail prior to assuming the watch. Step 4.22 of Procedure GP-005 was added as a corrective action to a similar event which occurred at Shearon Harris in 1989. This step states:

> "During power ascension, all indications of reactor power level should be monitored and compared. Periodically, indications such as core delta T and turbine first-stage pressure should be compared to NI indications. If all indications do not agree within 5 percent, Reactor Power should be stabilized and an OST-010 performed."

Application of this precaution would have identified the Power Range Nuclear Instrumentation mismatch earlier in the startup. Since this need to compare indicated power with other indications of power was not duplicated in the body of the procedure, this precaution was not brought to the operators' attention.

Immediately prior to the power escalation, the on-shift crew was relieved with the turbine latched and rolling. Precaution Step 4.15 of Procedure GP-005, limits the time that the turbine can be operated below 520 rpm. Concerns with violating this precaution were cited by at least one watchstander as providing an impetus for relieving the watch and commencing the power escalation. The desire to accomplish this expeditiously may have contributed to the less than adequate pre-evolution preparation.

Finally, the team noted that poor communications contributed to the failure of the watch section to diagnose the power mismatch. Although the reactor engineer's E-mail memo, discussed previously, contained specific direction for the operators to stop the power ascension if the 20 percent current equivalent rod stop was achieved prior to the P-10 interlock, the memo was not routed through the Operations Manager. A copy was provided to the Operations Manager; however, he stated that he was not aware of its existence prior to the power ascension. A second example of a communications failure was evidenced when two operators stated that they had raised questions over the accuracy of the indicated power range after power was stabilized at 20 percent. These concerns resulted from inconsistencies in turbine first stage pressure and indicated neutron power were made prior to the questioning of the net Mega Watts electric reading. However, these observations were not communicated to the entire crew for resolution and were not reflected in the operator logs.

The team concluded that sufficient information was available to control room operators to permit diagnosis of a deviation in indicated power and actual power prior to exceeding an actual power level of 20 percent. An analysis of control room instrument traces and plant computer records by the team revealed that several instruments indicated that the operators should have questioned the Power Range Nuclear Instrumentation readings. Further, these indications were available to the operators prior to exceeding an actual core power level of 20 percent. Specifically, loop delta Ts, turbine first stage pressure, and one of the Regulatory Guide 1.97 wide range power level instruments, all provided clear indication that actual power was greater than that indicated by the Power Range Nuclear Instrumentations.

The team concluded that watchstander distractions during the power escalation contributed to the failure to detect the Power Range Nuclear Instrumentation miscalibration. These distractions were primarily the result of the following: a focus by key watchstanders on preventing a reactor trip that overrode maintaining adequate oversight; difficulties in controlling certain plant parameters; and equipment malfunctions.

The team also concluded that the small magnitude of the Moderator Temperature Coefficient during the startup increased the necessity for reactivity control through frequent rod motion. Additionally, these power fluctuations added to the difficulty in controlling steam generator water levels.

Contributing factors included distraction of watchstanders, an inadequate pre-evolution brief, watch relief with the plant in an other than stable condition, and poor communications.

3.3.2 Operator Training

In 1989, the licensee's plant Harris experienced a similar event in which the Power Range Nuclear Instrumentations were discovered to be miscalibrated during a reactor startup. Industry Document SOER-90-003 described this event and corrective actions. The training on this industry event, at another facility owned by the same licensee, did not adequately prevent its recurrence at Robinson. The lessons learned from this event were covered only once during requalification training after the Harris event. It was not incorporated into any of the requalification training given after the first year. The training method did not adequately reflect the problems encountered in the Harris event; for example, the simulator scenarios did not challenge the operators with a Power Range Nuclear Instrument that was indicating low during startup conditions. The scenario had the operating crew detecting the inaccurate indication at 90 percent power by doing a procedurally required calorimetric. This did not reinforce the concept of monitoring diverse indications of power during a startup.

The initial license training program contains a lesson plan with an excellent description of the Power Range Nuclear Instrumentation miscalibration event at Harris. This was not used in requalification training. The only operator on shift who had recently been licensed, received this training but could not recall the details of it.

Several operators stated that the startup training did not adequately reflect the actual plant startup. The crew members participated in about four hours of simulator startup training. This startup contained no malfunctions. The simulator feedwater controls and instrumentation respond with such precision that the operators get little training on what is experienced in an actual startup. There were no distractions which would have required crew prioritization or coordinated oversight. Additionally. the startup training did not mention the new low leakage core. This would have been helpful in informing the operators of the expected response by the Nuclear Instrumentations.

Another contributing factor identified from the operator training was the lack of management involvement in the training. There was no Operations Department interface to relay their expectations to the control room operators.

3.3.3 Discussion of the Bypass of Intermediate Range Instrument Nuclear Instrumentation-36 High Flux Trip.

At 8:15 am on November 14, 1993, the level trip switch on intermediate range instrument NI-36 was placed in the bypass position to defeat the intermediate range high flux trip. This trip occurs at a nominal intermediate range current equivalent to 25 percent reactor power. The trip is not considered in the plant's safety analysis and is not required by technical specifications. When the trip was bypassed, NI-36 was indicating approximately 7.6×10^{-5} amps with the trip set at 1.3×10^{-4} amps. This action deviated from Startup Procedure GP-005 which required that the trips be defeated using the Intermediate Range "A" and "B" Logic Trip Defeat push-buttons on the gauge board. This can only be accomplished when the P-10 interlock is satisfied; i.e., 2 of 4 power range instruments indicated greater than 10 percent. When questioned on the appropriateness of this action, the watchstander stated that this evolution had been pre-briefed with the shift supervisor, that the P-10 interlock was satisfied prior to the action, and that the action was taken due to the tolerances assigned to the trip setpoint. As described by the watchstander and others, a trip on intermediate range high flux had occurred during a previous startup with little warning.

From a review of plant computer printouts, the team noted that at the time the NI-36 high flux trip was logged as defeated, the P-10 interlock was satisfied. The team also noted that the intermediate range high flux trips were correctly defeated in accordance with Procedure GP-005 approximately 1 minute later at 8:16 a.m. Based on their review, the team concluded that though the action was not in accordance with the startup procedure, the safety significance of this deviation was minimal.

3.3.4 AIT Findings and Conclusions

- Operators had sufficient indications to detect the difference between power range indications and actual reactor power.
- The crew's focus on trip prevention overrode maintaining adequate oversight.
- The operating crew did not trust their Intermediate Range Nuclear Instrumentation indications.
- There was not a pre-evolution brief to adequately emphasize precautions or expectations of the operating crew.

- Start up training did not reflect the actual plant start up.

- Training on the Harris event was not effective in preventing the occurrence of a similar event at Robinson.

- The potential consequences of this event were minimized since the initial goal of the power ascension was to stop at about 30 percent. The power ascension was stopped early with readings of the power range nuclear instruments indicating about 20 percent, but documentation of other readings at that time found the power to have been actually at 30.3 percent.

- Startup Procedure GP-005 did not prevent reoccurrence of the Harris event. Although a precaution to monitor diverse indications of power during power ascension was added, this was not read or implemented by the crew during this start up. There were no expected values for Mega Watts electric, delta T, or turbine impulse pressure listed in the procedure to flag problems at 10 percent and 20 percent power. This lack of guidance was a significant contributor.

- Management did not make their expectations clear as to control room watchstander duties and responsibilities.

4.0 CORE NEUTRON FLUX ANOMALIES

4.1 Assessment of root cause and safety significance of the core neutron flux anomalies with regard to fuel and technical specification limits.

Early in the core design turnover process from the fuel vendor to the licensee, the licensee's Nuclear Fuel Services group noted some computer input discrepancies (from the INCORE Code) which the fuel vendor then addressed. After the core was delivered, the final approval was then given by the licensee's Nuclear Fuel Services for the site to load the core and conduct startup physics and power ascension tests. Two in-core maps were taken at 30 percent power between November 14 and 16, 1993. The first map was number 698, and the latter one was number 699.

An error in different computer core design data (using the PDQ Code) was detected after the 30 percent power in-core flux maps had been performed. The fuel vendor's PDQ computer input deck that was part of the in-core analysis conducted on November 16-18, 1993, did not include two items:

- ITEM 1. The gadolinium rod overlays for the latest core reload batch (This computed higher predicted individual assembly powers in the gadolinium rods than if the gadolinium overlays had been used, and consequently other rods assembly powers were predicted lower.)

- ITEM 2. The six misconstructed asymmetrical gadolinium fuel assemblies (This item was, of course, not known at the time.)

A new computer in-core analysis was rerun the first week of December with the PDQ computer input corrections.

- The new computer input deck (with ITEM 1 fixed and presuming the core was loaded as-designed with properly configured assemblies) was run by the licensee's Nuclear Fuel Services for the number 698 and 699 maps taken at 30 percent power. Results were calculated as follows:

  - The new computer data calculated the predicted relative assembly power for the original expected design.
  - The in-core maps, numbers 698 and 699, showed the as-measured condition of the core at 30 percent power and calculated the as-measured relative assembly power.
  - The differences were then computed on an assembly by assembly basis and showed what should have been the results on November 16, 1993.

  This would have been the information available to the site and the licensee's Nuclear Fuel Services to decide if the as-measured core contained any anomalies that were significant, and whether the misconstructed assemblies would have been detected.

This analysis found that in-core map indications would have been present and, if the original computer data had been correct, would have led engineering to detect the misconstructed assemblies when the first maps were analyzed.

- Another computer input deck (with ITEM 1 fixed and the core loaded as-built, with the six misconstructed assemblies) was run by the licensee's Nuclear Fuel Services for the number 698 and 699 maps. This calculated the actual November 16, 1993, F-delta H and showed that the technical specification limit was exceeded by less than 0.5 percent.

  - The new computer data calculated the predicted relative assembly power in the as-built core.
  - The in-core maps, numbers 698 and 699, showed the as-measured condition of the core at 30 percent power and calculated the as-measured relative assembly power.
  - The differences were then computed on an assembly by assembly basis and simulated what could have been the results on November 16, 1993, if the core had been loaded correctly. This allowed the in-core map to be analyzed for any other discrepancies.

No other anomalies were observed in this analysis and discrepancies are not expected to be seen after the core is reloaded with the misconstructed assemblies in the proper locations.

4.1.1 AIT Findings and Conclusions

The team reviewed the Cycle 16 analysis, that was run with the correct design parameters, and found:

- The F-delta H technical specification limit was exceeded by less than 0.5 percent on one in-core map and the other map did not show a technical specification violation. This shows that the limit may have been exceeded by a small amount.

- The core radial tilt resulting from the misconstructed fuel assemblies was observable once the corrections were made to the computer data.

- The Cycle 16 fuel had operated well within the Departure from Nuclear Boiling limits and no damage to the fuel should have occurred. Reactor coolant chemistry data analysis also showed no fuel damage.

4.2 Assessment of the cause and extent of the fuel manufacturing errors at the Siemens Power Corporation-Nuclear Division fuel manufacturing facility.

On November 18, 1993, during cycle-16 plant start-up, it was determined that a manufacturing error had occurred and six misconstructed fuel assemblies had been installed in the Robinson core. The fuel assemblies had been built 90 degrees out of the correct orientation because incorrect load map drawing information had been entered into the manufacturing computer system. Two subsequent Quality Control overchecks failed to detect the error.

Fuel assembly manufacturing was controlled by the Bundle Assembly Data Logger computer system. The computer program compiled information associated with the fuel assemblies and provided technicians with manufacturing control instructions which indicated which rods to place in which position of the fuel assembly and in what sequence to do so.

Fuel assembly information was loaded into the computer program by a Production Control Clerk (clerk) who performed this function to assist the Production Control Specialist (specialist) who typically performed the task. The clerk used the specialist's identification and password to access the computer to perform this task. The fuel vendor indicated that the common use of the identification and password was not prohibited and that separate identification was not set up specifically for the clerk because he performed a variety of tasks. The process of entering the fuel assembly information into the computer program was not specified by procedure, and the clerk had only an informal document available for guidance. In addition, the clerk performed the task of loading the fuel assembly information into the computer program at a remote computer terminal, between other employees' work stations, with little work space to accommodate the required documentation.

The computer program was configured so that a specific set of information (header information) appeared at the top of the screen. The header information included the fuel assembly (bundle) item (part) number and drawing number, the load map item (part) number and drawing number, the manufacturing order number, and the project title. When entering the information for fuel assembly item number 140148, the clerk entered an incorrect load map drawing number 308181 (the correct load map drawing number was 308180 for fuel assembly item number 140148). When entering the information for fuel assembly item number 140150 the clerk entered the incorrect load map drawing number 308180 (the correct load map drawing number was 308181 for fuel assembly item number 140150).

The fuel vendor determined that the clerk had worked on fuel assemblies 140148 and 140150 during the same session at the computer program computer terminal and that it appeared that the load maps and the attached insertion sequences were swapped between the two packages.

The computer then provided a series of prompts to allow the clerk to enter the location of the fuel rods within the fuel assembly. The documents used to load in the fuel assembly information included the parts list (a reviewed and approved design document), the load map drawing (a reviewed and approved design document), and the insertion sequence (an informal document which indicated the order in which the assembly table placed fuel rods into the fuel assembly, a manufacturing document). The clerk defined a set of "find numbers" used to identify rod types and then entered the insertion sequence using the find numbers. This information was obtained from the load maps and insertion sequences which were incorrectly specified for fuel assembly numbers 140148 and 140150. As a result, the computer program was loaded with design information and manufacturing instructions which would place the fuel rods in incorrect positions when the fuel assemblies were manufactured.

After entry of the fuel assembly information was completed, the computer program produced a bundle proof map (the header information, a matrix of find numbers representing the fuel assembly, and the find number definitions) for each fuel assembly. The bundle proof maps for fuel assembly numbers 140148 and 140150 were verified by Quality Control Engineering by comparing the matrix of find numbers and the find number definitions to the load map drawing specified in the header information for each fuel assembly. However, the Quality Control Engineering person did not verify that the load map drawing number listed in the header information was correct (it was incorrect for fuel assemblies 140148 and 140150). The team reviewed the procedure governing the quality control activities, "Fuel Bundle Map Verification," Revision 0, dated August 3, 1990, and noted that it did not clearly specify the basis for the checks but only specified that the checks be made against a "hard copy."

Following completion of the manufacturing process, the computer program provided an as-built bundle map of each fuel assembly which listed the part number (or no load) which was located in each coordinate of the fuel assembly. A Quality Control Inspection Technician reviewed the as-built bundle maps for fuel assemblies 140148 and 140150 against the load map drawings which were specified in the computer program header information; however, the technician did not verify that the load map drawing number listed in the header information was correct (it was incorrect for fuel assemblies 140148 and 140150).

As a result of the error made in the Robinson fuel assemblies, the fuel vendor reviewed the as-built lists and records for ROB-13 (Robinson's Cycle 16 refueling) and the remaining assemblies in the Robinson core and approximately

1000 additional fuel assemblies the fuel vendor had previously manufactured. The fuel vendor determined that no other misconfigurations existed. The team reviewed the documentation of the ROB-13 review and determined that this method had credibility and that the extent of the problem appeared to apply only to the six Robinson fuel assemblies in question.

4.2.1 AIT Findings and Conclusions

- The team concluded that the control of design information (i.e., the required location of the fuel rods within the fuel assemblies) as it was translated into the computer system was inadequate; that the level of responsibility, accountability, supervision, and review associated with this critical task was inadequate; that the performance of the quality control overchecks of the computer program information was inadequate; and that the procedures used to govern the activities were inadequate.

- On November 22, the fuel vendor stated that they had verified 100 percent of the current core load. They compared the as-built documentation to the core design documents and stated that no other error was made. the licensee independently verified this documentation. The team verified that this method had credibility.

  The failure of Quality Control to compare as-built information to design documents was the basic flaw that resulted in the misassembly of the fuel.

4.3 Assessment of the extent and effectiveness of fuel assembly verification at the Siemens Power Corporation-Nuclear Division.

The team determined that the fuel vendor used several methods to maintain accountability of the fuel assemblies and their subcomponents. The methods included computer-readable bar codes, man-readable serial numbers, hard copy travellers, and Quality Control overchecks. At the initial stage of fuel rod manufacture, a serial number was engraved on the lower end cap in the form of a computer-readable bar code and man-readable number. This serial number was entered into the Rod Serialization System computer system which tracked the fuel rod through the assembly process, from manufacture of the lower end caps through storage of the completed fuel rods.

The lower end cap was welded to the tube of cladding and the lower end cap serial number was scanned to associate the lower end cap serial number with the manufacturing order number, the fuel rod group number, the fuel rod part number, and the clad part number. The fuel rod was further assembled by loading the fuel pellets, the load spring, out-gassing the fuel rod and welding the upper end cap to the fuel rod. Following assembly, a leak check, a uniformity check, a through rod x-ray, and a final inspection for color, straightness, length, and weld quality was performed. During each step in the process the lower end cap serial number was computer scanned at the work station to maintain accountability for completion of the particular portion of the fuel rod manufacture. At the end of the fabrication process, the fuel rods were placed in storage bins near the fuel assembly manufacturing area.

The Bundle Assembly Data Logger (the computer program) computer system then tracked the fabrication of the fuel assembly from removal of the fuels rods from storage through completion of the assembly process.

Fuel assembly manufacturing occurred when a manufacturing order created a demand and the Quality Control checks had been performed of the fuel assembly information in the computer program, which had been entered by the clerk, and Quality Control Engineering had electronically enabled the computer program to allow the manufacture of a fuel assembly.

Fuel rods were moved from storage onto the order picker by an elevator. The order picker had 12 trays divisible by 3 to allow for a total of 36 types of rods to be on the machine at a given time. The computer system was tied to the order picker which provided a signal light to indicate from which section rods were to be removed in accordance with the insertion sequence. As the rods were removed from the order picker, the lower end caps were scanned into the computer program and the order picker decremented the count of the rods in the applicable section.

The fuel rods were moved from the order picker to the loading section of the insertion table. The fuel rods were scanned as they were placed on the downward slope of the insertion table in the order of the insertion sequence. The fuel rods were picked up from the slope of the insertion table by a set of notched wheels, scanned while in the wheel to verify the insertion sequence, and dumped into a feed trough. The assembler then moved to the proper "x-y" coordinates, based on the insertion sequence, and the fuel rods were pushed into the specified "x-y" coordinates of the fuel assembly located on the assembler table where the fuel rods were then fixed into place.

After completion of the manufacture of the fuel assembly, the computer program printed out the as-built bundle map, a sequential list of the fuel assembly "x-y" coordinates and the part numbers and serial numbers of the items which filled the coordinates (such as fuel rods). The Quality Control Inspection Technician compared the as-built bundle map to the load map specified in the computer program header data to verify that the assembly had been correctly manufactured.

4.3.1 AIT Findings and Conclusions

The team concluded that the fuel vendor appeared to have an effective program for assembly verification in most areas. However, the team did conclude that the fuel vendor performance in the area of entering design information (the load map fuel assembly pattern) into the computer program computer system and the subsequent Quality Control overchecks of this process were less than adequate (See paragraph 4.2 of this report). In addition, the fuel vendor indicated that the level of complexity of the fuel assemblies being manufactured had increased since the manufacturing system had been implemented in 1984. The team determined that the manufacturing machines and computer systems involved did not appear to have been worked beyond capacity; however, there did appear to have been an increasing level of complexity of the information required to be manipulated and verified by the personnel involved in the manufacturing process.

4.4 Assessment of the extent and effectiveness of fuel verification at the site.

Fuel verification at the site consisted of a visual inspection upon receipt. This included a visual inspection of the assembly externals and the assembly serial number. This type of inspection is similar to the industry standard.

4.4.1 AIT Conclusion

No method is reasonably available at the site for more detailed verification.

4.5 Assessment of the Siemens Power Corporation-Nuclear Division analysis of the core neutron flux anomalies.

The team reviewed and evaluated the fuel vendor's performance and response to the observed core flux anomalies at Robinson during power ascension testing. The fuel vendor had no role in the Robinson miscalibration of the excore nuclear power range instrumentation. Their only involvement was to provide confirming data for the re-calibration procedure, after the incident, including reasonable weighing factors for edge bundles adjacent to the excore Nuclear Instruments.

The team assessed the fuel vendor's analysis of the Robinson core power tilt and bundle power anomalies. These were observed at the 30 percent power level testing, after processing of the in-core flux map measurements, with the licensee version of the Westinghouse INCORE program using the fuel vendor provided input files. An interactive licensee and fuel vendor analysis of the 30 percent power measurement results, including rerunning the in-core map, led to suspension of power ascension until the anomalies could be resolved. The fuel vendor performed an independent analysis of the in-core measurements (with their INPAX-W program), confirming that the Cycle-16 core was not performing as designed. The fuel vendor, after an extensive Quality Assurance record review, then reported their discovery that six fuel bundles had been misbuilt, which had resulted in a reload core misconfiguration. The fuel vendor provided revised INCORE input decks for the as-built condition, allowing the core power distribution to be evaluated against the design peaking factors. However, the fuel vendor later discovered an INCORE input error for all fresh, burnable poison bundle types, after noting an in-core flux anomaly in other than the misbuilt bundles.

4.5.1 AIT Findings and Conclusions.

- The team determined that the fuel vendor's Pressurized Water Reactor Nuclear Engineering (NE) support, after the observed INCORE anomaly indication, was appropriate and adequate in confirming the core mis-loading. However, the initial licensee discovery and subsequent fuel vendor correction of earlier INCORE input file errors should have triggered a complete deck review by the fuel vendor and could have alerted both the licensee and the fuel vendor management to potential problems in the core design process.

- The team determined that deficiencies in the fuel vendor's analysis and verification procedures caused the input errors in the in-core flux mapping computer program (INCORE). Also, the fuel vendor's regeneration of INCORE computer input for the as-built core should have included a complete deck review, which might have discovered the computer input errors earlier. The fuel vendor's response after their INCORE computer input error discovery was appropriate and supplied the proper level of support.

4.6 Assessment of the licensee's oversight of Siemens Power Corporation-Nuclear Division's fuel analysis and Quality Assurance programs.

The team reviewed selected areas from the fuel vendor reload design analyses from Robinson cycles-14, -15, and -16 core reloads. The fuel vendor's core reload design activities began approximately 18 months prior to fuel delivery upon receipt of the Tentative Scheduled Delivery Date - notification from the licensee. This includes the projected end-of-cycle performance of the current cycle, and the estimated energy requirements for the target cycle. Based on this notification and discussions with the licensee, the fuel vendor's Nuclear Engineering provided the licensee's Nuclear Fuel Services with a preliminary fuel bundle and core design. This tentative design, including the number of bundles and rod types, was also provided to the fuel vendor's Product Mechanical Engineering for mechanical design and material requirements development.

Approximately 12 months prior to fuel delivery, the licensee's Nuclear Fuel Services provided the Final Scheduled Delivery Date - notification to the fuel vendor, along with the final cycle energy requirement and an upper and lower energy generation window for the current end-of-cycle. Based on this data, the fuel vendor provided the final fuel bundle and core design to the licensee and delivered a Characteristic Specifications document to their Product Mechanical Engineering for the development of the detailed parts list. The licensee reviewed the design and provided approval. At this point, the reload core design was considered final; however, the fuel vendor documentation had not undergone Quality Assurance review and approval.

For the Robinson cycle-16 core reload, the original "final" design utilized a 48-bundle split-enrichment batch to achieve a 430 equivalent full power day operating cycle. At the licensee's request, a further design change was developed by the licensee and the fuel vendor (November 1992) to allow the 48-bundle reload batch to be reduced to 44 bundles. This was achieved by going to a low leakage loading core and utilizing a single enrichment bundle design with more burnable poison fuel types. At this time, the effect of the low leakage loading pattern to reduce excore detector response was noted by the licensee.

Based on the changed design, the fuel vendor's Nuclear Engineering revised the Characteristic Specifications documentation and provided updated load map data to Product Mechanical Engineering for development of the final parts list and bundle ID core maps. Following their standard Quality Assurance procedure, the fuel vendor then began the preparation and review process for their formal documentation packages and provided them to the licensee during the four months prior to fuel delivery.

The Robinson cycle-16 Reload Batch Design Report, signifying the fuel vendor Quality Assurance review and approval of the reload design, was issued to the licensee in April 1993. The Safety Analysis Report was issued in August 1993 and the Startup and Operations Report was issued in September 1993. The INCORE monitoring input file (deck) was officially transmitted by letter with a diskette in October 1993. Several corrections were noted by the licensee and the fuel vendor provided revisions to the INCORE input file between the end-of-cycle 15 shutdown and before Cycle 16 startup.

The team's assessment of the licensee activities are as follows:

- There was less than adequate licensee oversight of the fuel vendor cycle-16 core reload design implementation and verification activities and of the quality assurance procedures concerning the calculation notebooks.

- The licensee conducted independent calculation reviews of the fuel vendor reload design parameters at their Raleigh offices and primarily judged design adequacy based on agreement between the two design models.

- The licensee did not conduct onsite audits of the fuel vendor Nuclear Engineering and Product Mechanical Engineering design activities and interfaces for the cycle-16 core reload.

- The last known Robinson audit that covered neutronic areas was during the cycle-12 reload design period.

- The licensee conducted in-house reviews of the fuel vendor generated INCORE computer input file; first by processing the file through a data curve plotting routine, and then by making test runs using predicted in-core flux measurement data. For the cycle-16 deck, this process revealed incomplete data and several errors in the initial fuel vendor transmittal. Corrective actions did not include a broader review of the total reload design process.

4.6.1 AIT Findings and Conclusions

The team determined that the licensee's oversight process was inadequate because the licensee's Nuclear Fuel Section:

- failed to review or observe any of the fuel vendor's fuel bundle assembly manufacturing activities (due to fabrication schedule changes that occurred during Nuclear Fuel Service's surveillance activities); and

- failed to compare the fuel vendor's fuel bundle assembly records to the characteristic specifications for Robinson's cycle-16 fuel load. The licensee's Independent Assessment Team investigation of the Robinson cycle-16 fuel and core reloading problems also concluded that its oversight of the fuel vendor's fuel manufacturing activities was less that adequate.

5.0 BROKEN FUEL INSPECTION TOOL DURING REFUELING

5.1 Determination of the Root Cause of the Broken Fuel Inspection Tool Event.

5.1.1 From the inspection on-site.

On October 11, 1993, licensee personnel discovered that the control rod for fuel assembly U-24 would not completely insert into the assembly. The control rod stopped with approximately two feet of full insertion remaining. An inspection of selected fuel assemblies in the spent fuel pool was in progress by the licensee's fuel vendor, Siemens Power Corporation, to measure fuel assembly and fuel rod lengths. These measurements had routinely been made at other nuclear facilities using similar tools. Further investigation and conversation between the licensee and the fuel vendor revealed that loose parts from a damaged fuel inspection tool had been dropped into a control rod guide tube of assembly U-24.

The team reviewed the fuel vendor site activity log for the conduct of these measurements and also reviewed the licensee's log of these activities to determine and verify the sequence of events. Also the vendor's incident review board report and licensee's self assessment report were reviewed. The team discussed these activities with licensee oversight personnel and the fuel vendor team leader who was responsible for the performance of the fuel measurements. Both the licensee's and the vendor's log books were very brief and did not provide any detailed information. Neither party had developed any standards for these logs.

Special Procedure SP-1258, "Fuel Assembly Inspection and Repair," provided guidance for the conduct of these measurements. This procedure was reviewed by the team. As part of the procedure guidance for the fuel inspection, the fuel assembly upper tie plate was removed and a reference plate installed. This reference plate was held in position by the use of three expandable anchors which inserted into three guide tube locations. The anchors were hand tightened to secure the plate into position. Although the tool had been used before, new anchors had recently been fabricated and installed for this inspection.

From a review of the logs and discussions with personnel, the team developed the following sequence of events. The fuel vendor personnel measured assembly S-15H on October 6, 1993. Following this inspection, the tool (including the reference plate) was removed from the assembly and placed in a temporary three foot storage area on the side of the spent fuel pool. On October 9 at approximately 9:00 pm, the tool was utilized again on assembly U-24. The fuel vendor personnel noted that only two of the expandable anchors engaged and no resistance was noticed when tightening the third anchor. This information was not included in the fuel vendor site activity log or communicated to licensee personnel. The fuel vendor personnel decided the tool could still function in this condition and continued using the tool. Following the measurements on assembly U-24, the tool was used again on assembly U-23. Fuel assembly measurements were completed at 7:20 am on October 10, 1993. On October 11, 1993, at approximately 2:00 am, the tool was removed from the spent fuel pool and the fuel vendor personnel noted damage to the reference plate and missing parts from the expandable anchor. Specifically, the missing parts included an expander nut, expander tube, two roll pins, and a portion of the clamp shaft (see figure X). Damage to the clamp shaft was also observed which appeared to be slightly bent. This information was also not recorded in the fuel vendor site activity log nor communicated to licensee personnel. Later that same day at approximately 9:00 pm, licensee personnel experienced difficulty while inserting the control rod for assembly U-24 at which time the fuel vendor personnel reported the missing parts from the tool to licensee personnel. The licensee performed a visual examination of the control rod and found the expander tube lodged on the tip of a control rod finger. The licensee considers the other missing parts to likewise be in the same guide tube where the expander tube was found based on the construction of the anchor which would not provide any support for the tube or roll pins once the expander nut was removed. Licensee personnel examined the guide tube with a plug gage and found blockage evident near the dash pot region of the guide tube. This also indicated that the expander nut was still in the guide tube.

Due to the damage noted on the clamp shaft, the fuel vendor believes that the tool was damaged by an impact from another tool while placed in temporary storage in the spent fuel pool for the approximate four days between tool usage. The small, three foot area allocated was congested with many tools (approximately 13). Several other tools were moved to this same storage area during the time frame involved. The licensee's investigation did not reach a conclusion on how the tool was damaged. Due to the small amount of force which would be required to overtighten and break the anchor, the team concluded that the tool could have been damaged either by an impact while in the storage area or from overtightening.

The team discussed with the vendor the delay in vendor personnel reporting the missing tool parts to the licensee. Both the licensee and the vendor incident review board determined that this omission was not a deliberate act but rather a significant error in judgement by vendor personnel. Past experience of the vendor personnel indicated that at other facilities where loose parts were dropped in the spent fuel pools, licensee personnel had been informed shortly afterwards. The vendor team had considered that the lost material could be resolved at a later time and the missing parts were thought to be on the bottom of the pool and not in a fuel assembly. Although the vendor did not have a procedure on foreign material exclusion, vendor personnel were trained by the licensee on the requirements of Procedure PLP-047, Foreign Material Exclusion Area Program. Access to the foreign material exclusion area was strictly controlled by licensee personnel and logs were required to document entering the area or bringing in material. The vendor was further requested by the licensee to remove debris from several fuel assemblies prior to fuel load in the core indicating the sensitivity of this subject. The team concluded that the fuel vendor personnel should have been aware of the importance that a loose part in the spent fuel pool would have. The investigative reports also mentioned the poor fitness for duty of personnel during this time frame due to illness and being physically tired following several 12 hour shifts of work. This poor physical condition could have contributed to the poor judgement of contractor personnel who failed to promptly notify licensee personnel of the missing parts/damaged tool. Due to these facts, and discussions with the vendor team leader, the team agreed with the vendor's determination that the failure to promptly report the missing parts was not a deliberate act.

The licensee performed an engineering evaluation for the continued use of the fuel assembly with the loose parts inside the guide tube. This evaluation concluded that this action would be acceptable based upon chemical, thermal, and mechanical compatibility of the loose parts with the rest of the assembly. Therefore, licensee personnel installed a thimble plug over the guide tubes for this assembly. However, as a result of this action, eight fuel assemblies had to be repositioned in the core loading to substitute a rodded fuel assembly for the plugged assembly.

5.1.2 Determination of the root cause of the broken fuel inspection tool event from the fuel vendor Inspection.

Siemens Power Corporation-Nuclear Division Fuel Services department conducted fuel examinations in the spent fuel pool during Robinson's cycle-16 refueling outage. As part of its fuel examinations, the fuel vendor's site team measured the length of the assembly and the fuel rods of three fuel assemblies that were examined in the following order: S-15H, U-24, and U-23. The length measurements require removal of the upper tie plate from the fuel assembly followed by the attachment of a reference length plate. The reference length plate is designed to be attached to the fuel assembly at three guide tube locations using a remotely activated expandable anchor inserted in each guide tube. The expandable anchor consist of a clamp shaft with an expander nut (collet) on each end of a slotted expander tube and two roll pins (one is a backup) inserted through the clamp shaft below the bottom expander nut. The expander nuts are drawn into a slotted expander tube by remotely turning the clamp shaft.

The length measurements of fuel assembly S-15H were completed without incident on October 6, 1993. On October 10, 1993, during the attachment of the reference length plate on fuel assembly U-24, the expandable anchor inserted in guide tube location E-11 failed to tighten within the guide tube. Siemens Power Corporation-Nuclear Division site team determined that the length measurements and examinations of fuel assembly U-24 could be completed without utilizing the malfunctioning expandable anchor. The length measurements of fuel assembly U-23 were also completed on October 10, 1993, using only two functioning expandable anchors. On October 11, 1993, as part of its general pack-up activity, the fuel vendor site team removed the reference length plate from the spent fuel pool for the first time since the fuel assembly length measurement examinations began on October 5, 1993. Once the reference length plate was removed from the spent fuel pool, the fuel vendor site team discovered that the malfunctioning expandable anchor clamp shaft was broken, bent, and missing the following parts:

- lower expander nut
- expander tube
- two roll pins
- a portion of the clamp shaft

Upon discovery of the broken expander anchor at Robinson, Siemens Power Corporation-Nuclear Division Fuel Services Engineering staff destructively tested an identical expandable anchor and found that the clamp shaft failed in the region of the uppermost 0.1574-cm (0.062-inch) diameter roll pin with an applied torsional load of approximately 4.324-N (35-lbin or 0.972-lbf).

In Siemens Power Corporation-Nuclear Division Incident Review Board report EMF-93-195(P), "H. B. Robinson Fuel Examination, September 21-October 14, 1993, Investigation of Failure to Report Loose Parts," issued on November 5, 1993, the fuel vendor stated, in part, that the issues summarized below were identified by the Incident Review Board as the root cause/causal factors for the loose parts event at Robinson:

- The applicable Standard Operating Procedure, Fuel Performance, EMF-P71,129, "Fuel Rod and Assembly Length," Revision 0, March 27, 1992, did not reference the reference length plate Drawing No. ANF-306,200, "Rod Length Measuring Tool," Revision 0, September 4, 1987.

- The fuel vendor's Fuel Services equipment technicians prepared the reference length plate tool and shipped it to Robinson without referring to the reference length plate drawing to verify its configuration. Siemens Power Corporation-Nuclear Division failure to verify the tool's configuration resulted in placing a tool in service that was in less than adequate condition because the reference length plate was designed with four guide pins and it was shipped to Robinson with only three. The guide pins were designed to aid in orienting the tool to the fuel assembly by inserting them into the guide tubes. In addition to its orienting function, each guide pin also served to protect the expandable anchors.

- The fuel vendor concluded that the expandable anchor design was poor and found no documented engineering review of the expandable anchor design. According to the fuel vendor, the design of the expandable anchor is inconsistent with its design philosophy which dictates that tool failure will not result in loose parts.

- Siemens Power Corporation-Nuclear Division Incident Review Board concluded that the reference length plate and expandable anchor were most probably struck from the side and that the impact was likely from another tool or hardware in the spent fuel pool. The fuel vendor added that the difficult handling of heavy tools, crowded conditions, and inadequate preparation of the work area contributed to the potential for such an impact. Additionally, the fuel vendor noted that the absence of the forth guide pin reduced the physical protection of the expandable anchor from external damage.

On November 30, 1993, the NRC's team at the fuel vendor's facility inspected the reference length plate used at Robinson by the fuel vendor's site team. The broken expandable anchor clamp shaft and upper expander nut were found in their original as-built location on the reference length plate.

The fuel vendor agreed to perform additional examinations of the fracture surfaces of the broken clamp shaft and compare its appearance to the fracture surfaces of the expandable anchor clamp shaft from the destructively tested expandable anchor. The additional examinations were performed on the two clamp shafts using both low magnification micrographs taken with an optical stereo-microscope and scanning electron micrograph mosaics of the fracture surfaces.

From the results of these examinations, the team determined that the root cause of the broken expandable anchor was a ductile overload of the clamp shaft at the upper roll pin location that was induced through multiple incremental overtorquing events. Siemens Power Corporation-Nuclear Division analysis of the fracture surfaces and the results of these examinations (documented in DTP:93:033, "Analysis of Fuel Services Component Failure," dated December 3, 1993) reached the same conclusion. The team also determined that the location where the expandable anchor overtorquing events occurred (including the event that resulted in the 2° bend of the clamp shaft) is indeterminate in that these events may have occurred during functional testing of the reference length plate in the fuel vendor's mock-up pool, prior to its shipment to Robinson, or during the fuel assembly examinations performed at Robinson.

The team also determined that the fuel vendor's original root cause analysis of the failed expandable anchor (documented in Incident Review Board report EMF-93-195(P), "H. B. Robinson Fuel Examination, September 21-October 14, 1993, Investigation of Failure to Report Loose Parts," issued on November 5, 1993) was less than adequate because it did not determine the mechanical failure mechanism of the broken clamp shaft.

5.1.3 AIT Findings and Conclusions

During fuel preparation for core load, loose parts had been found in a fuel assembly as a result of a fuel inspection reference tool breaking. The fuel vendor had been conducting fuel inspections in the Robinson spent fuel pit.

The root cause of the broken fuel inspection tool was attributed to the fuel vendor design control problems and inadequate licensee oversight. The current fuel vendor design control system had been set up in 1988 with no requirement to use this methodology on items constructed after this date but built to an earlier design. This particular design was completed in 1987 and the tool constructed in 1992 and refitted in 1993.

The licensee's analysis of the small, unrecovered parts found that they were confined to the fuel assembly guide tube and presented no future threat to fuel integrity. The team agreed with this finding.

5.2 Effectiveness of Licensee Oversight of Contractor Fuel Handling Activities

The team reviewed the involvement of licensee personnel during the fuel measurements performed by the fuel vendor. This item was discussed with the licensee and contractor personnel involved. Also the team reviewed the licensee's assessment report.

According to licensee plant personnel, oversight of this activity included 24 hour coverage using 2 12-hour shift rotations. Licensee personnel were present on the fuel handling floor during the fuel assembly inspections. Only verbal guidance was provided to licensee personnel on their responsibilities for oversight functions which consisted of assuring that vendor personnel adhered to the licensee's procedures.

The team discussed this coverage with licensee personnel and was informed that in some cases, concurrent activities occurred in the pool. This interfered with direct oversight of the vendor activities. This occurred during the tool removal, when licensee personnel were concurrently inspecting fuel assemblies for debris. Licensee involvement in observing the video display for the debris inspection prevented the direct oversight of contractor personnel when the tool was removed.

The licensee's independent assessment report of this event concluded that licensee personnel were not always present during the fuel measurements. This contributed to the poor communication between the licensee and the vendor. The team could not determine from the logs how long licensee personnel were actually present for the fuel inspections.

Contractor activities for this fuel inspection were controlled by licensee Procedure SP-1258. This procedure included attachments with the vendor's procedure for the ultrasonic inspection, repair, and examination of fuel assemblies (EMF-1576). The completed procedure was reviewed by the team as well as the licensee's safety evaluation package of the procedure. The team noted that the attached vendor procedure contained references to detailed vendor procedures for the performance of the fuel rod and assembly length measurement (EMF-P71,129) and for the upper tie plate removal/reinstallation (ANF-P71,032). These two procedures were utilized by the vendor and involved partial disassembly of the fuel assemblies for the inspections including tie plate removal and the removal of fuel rods from the assembly. The team noted that these quality activities were not encompassed by the licensee's safety review of Procedure SP-1258.

The licensee provided training for the contract personnel on the implementing procedure and requirements of Procedure PLP-037. Procedure SP-1258, section 6.2, contained a specific precaution on the requirements for foreign material exclusion areas. The training consisted of handing out the procedures and allowing contractor personnel to "self study" the handouts. The contractor personnel acknowledged by signature the accomplishment of this training. The licensee's assessment report concluded that the contractor indoctrination for foreign material exclusion requirements was not effective.

The licensee's self-assessment report also identified that planning and coordination for this activity was inadequate based on insufficient space availability and availability of support services. This report also stated that the responsibilities of licensee personnel who provided the oversight function were not clearly defined.

The team noted from log book entries that both contractor and licensee personnel were physically tired and in some cases ill. This condition was not noted in subsequent licensee investigations.

5.2.1 AIT Findings and Conclusions

The team concluded that the licensee oversight of contractor activities was considered to be less than adequate; this was verified by interviews with personnel and review of logs. The contributing causes were:

- Poor planning and coordination of fuel inspection.
- Failure to identify and adjust staffing of personnel when conditions changed.
- Lack of clearly defined responsibilities.
- Poor safety review of vendor procedures for fuel inspection.

A review of the licensee's assessment report revealed their causes of the event to be: the poor planning and coordination of the fuel inspection activities, lack of licensee management to identify and adjust staffing of personnel when degradation of physical conditions was indicated, lack of clearly defined responsibilities for licensee personnel overseeing this activity, and a poor safety review of procedures utilized by the contractor in performing these inspections.

5.3 Assess the adequacy of Siemens Power Corporation-Nuclear Division Quality Assurance program for the manufacture of special fuel tools.

The reference length plate used at Robinson was designed in September 1987 and has been used several times at Robinson and Tihange in Belgium (both plants have 15 x 15 Pressurized Water Reactor fuel assemblies). However, the reference length plate design depicted on Drawing No. ANF-306,200, Revision 0, had not been reviewed in accordance with the fuel vendor's design review and design control measures established by the Quality Assurance program in 1988. Siemens Power Corporation-Nuclear Division failure to evaluate the reference length plate design in accordance with its Quality Assurance program is considered by the team to be a significant contributing factor in the loose parts event at Robinson.

In November 1990, Siemens Power Corporation-Nuclear Division Fuel Services Engineering apparently recognized the poor design of the expandable anchor that failed at Robinson. Drawing No. ANF-306,200, Revision 1, approved on

November 14, 1990, revised the expandable anchor design by (a) increasing the diameter of the clamp shaft, (b) threading the end of the clamp shaft and providing a threaded locking sleeve to retain the lower expander nut, and (c) inserting a roll pin through the threaded locking sleeve and shaft to ensure the locking sleeve would not loosen and back-off. Implementation of these design changes would have prevented the loose parts event at Robinson. However, the revised expandable anchor design was not manufactured or ever utilized by the fuel vendor. Siemens Power Corporation-Nuclear Division's failure to incorporate the enhanced expandable anchor design in the reference length plate was a missed opportunity that contributed to the loose parts event at Robinson.

Moreover, during its preparation of the reference length plate for shipment to Robinson, the fuel vendor missed another opportunity to prevent the loose parts event at Robinson. In September 1993, when the Fuel Services personnel retrieved the reference length plate from storage in preparation for its shipment to Robinson, it was discovered that one of the expandable anchors had missing parts (the lower expander nut, expander tube, and the roll pins). The fuel vendor responded to the discovery of missing parts by fabricating three new expandable anchors to the old, 1987 design. The fuel vendor failed to (a) determine what had happened to the missing parts (i.e., were the parts lost during the reference length plate's last usage, which was at Robinson during its cycle-15 refueling outage, and if so, can the missing parts be located), (b) fabricate the replacement expandable anchors in accordance with the revised 1990 design, and (c) perform an engineering evaluation of the newly fabricated expandable anchors' design in accordance with the requirements of the Quality Assurance program.

Subsequent to the team's identification of Siemens Power Corporation-Nuclear Division's less than adequate actions regarding the missing parts, the licensee at Robinson investigated the potential that the missing parts may have entered Robinson's spent fuel pool or core during Robinson's cycle-15 refueling outage. Although it was not possible to establish with certainty what happened to the missing parts, the licensee, on the basis of all available information, determined that the missing parts were not located in the Robinson spent fuel pool or core. The team reviewed the licensee's evaluation and accepted its conclusion.

5.3.1 AIT Findings and Conclusions

From its review of Siemens Power Corporation-Nuclear Division's Quality Assurance program, the team determined that for those fuel tools developed since the implementation of the Quality Assurance program, the Quality Assurance program appeared to be adequate. Siemens Power Corporation-Nuclear Division's Quality Assurance program for the manufacture of special fuel tools began in 1988 when the Fuel Services department first implemented Siemens Power Corporation-Nuclear Division's Quality Assurance Manual EMF-1.

However, for those fuel handling tools that were developed by the Fuel Services department before the implementation of Siemens Power Corporation-Nuclear Division's Quality Assurance program, Siemens Power Corporation-

Nuclear Division efforts to incorporate those tools into the Quality Assurance program's design review and design control measures appear to be less than adequate; as demonstrated by the reference length plate (Drawing No. ANF-306,200, "Rod Length Measuring Tool," Revision 0, September 4, 1987), described above.

5.4 Assessment of the effectiveness of Siemens Power Corporation-Nuclear Division's program for notifying licensees of known deficiencies in either hardware or services provided.

The team reviewed Siemens Power Corporation-Nuclear Division's program for review and notification to customers of identified deficiencies in hardware or services, with particularly interest in the three deficiencies identified during the Robinson event. The identified deficiencies were, (a) the presence of loose parts in the fuel pool, (b) the incorrect manufacturing of six fuel assemblies, and (c) the potential error in the generation of transport correction factors used in the generation of the INCORE code.

Siemens Power Corporation-Nuclear Division's program is controlled by Policy Guide 10.2, "Nuclear Safety Hazards Reporting," dated December 17, 1991. Policy Guide 10.2 was previously reviewed during a February 1992 NRC inspection at Siemens Power Corporation-Nuclear Division (see Inspection Report 99900081/92-01). During the 1992 inspection, the team found the Policy Guide contained all necessary requirements of 10 CFR Part 21. However, the 1992 inspection did identify an area of concern regarding the language relating to who has responsibility for performing evaluations. This concern was again raised to Siemens Power Corporation-Nuclear Division's management during this inspection.

As previously discussed, the fuel vendor Fuel Services personnel failed to report the presence of loose parts in the spent fuel pool to Robinson's staff in a timely manner. The loose parts resulted from a failure of the reference length plate expandable anchor. The licensee was not notified of the failure until Robinson personnel were unsuccessful in loading a control rod into fuel assembly U-24, approximately 20 hours after the fuel vendor site team first identified the parts as missing. The fuel vendor convened an Incident Review Board to investigate the circumstances surrounding the event and to determine 10 CFR Part 21 applicability. Incident Review Board report EMF-93-195(P), "Incident Review Board Report, H. B. Robinson Fuel Examination, September 21-October 14, 1993, Investigation of Failure to Report Loose Parts," concluded that there were no 10 CFR Part 21 implications from this event. The Incident Review Board report based its conclusion on justifying the use of assembly U-24 using a thimble plug device and moving its location in the core. The team questioned the adequacy of this conclusion in light of potential generic considerations regarding the use of the fuel tool at other facilities and the potential for loose parts escaping the guide tube and damaging other assemblies in the core. The fuel vendor provided the team with engineering evaluation, "H. B. Robinson - Project Variance Evaluation of a Piece of a Tool End in a Guide Tube of Assembly U-24," which was provided to the licensee in an October 13, 1993, letter (RAC:93:165). The engineering evaluation provided sufficient evidence that the loose parts in assembly U-24's guide tube (E-11) could not escape from the tube. The fuel vendor also informed the team that the tool was only used at Robinson and one foreign plant (Tihange in Belgium). Since Robinson is the only U.S. commercial facility that has used the tool, and they were notified approximately 20 hours after the fuel vendor site team identified the missing parts, further notification by the fuel vendor was unnecessary.

The problem with the incorrect manufacturing of six fuel assemblies was identified by the licensee during power distribution mapping at Robinson at approximately 30 percent power on November 18, 1993. The fuel vendor was notified of the power distribution anomalies and convened a Hazards Review Board the same day. The board concluded that the manufacturing error constituted a defect under 10 CFR Part 21 which could potentially pose a significant safety hazard and recommended that the condition be reported to the NRC. The fuel vendor further recommended that the licensee make the initial notification to the NRC. The licensee fulfilled the notification requirement later on November 18, 1993. Both Siemens Power Corporation-Nuclear Division and the licensee's actions regarding 10 CFR Part 21 were appropriate and in accordance with regulatory requirements.

The team further examined Siemens Power Corporation-Nuclear Division actions associated with the incorrect manufacturing of the fuel assemblies with regard to the identification of potential generic aspects. The fuel vendor formed an Incident Review Board to investigate this matter on November 19, 1993. The team reviewed draft Incident Review Board report EMF-93-209(P), "Incident Review Board Report, Misconfigured Fuel Assemblies at Robinson," dated December 1993. The report includes a review of reload records for Robinson, Susquehanna, Dresden, Grand Gulf, Peach Bottom, Kuosheng 2, WNP-2, Comanche Peak, and Laguna Verde to assure that each rod was located in the correct position in each assembly. The review encompassed approximately 1240 fuel assemblies. The fuel vendor did not identify any other mispositioned rods.

The team reviewed an internal memorandum of November 28, 1993, Siemens Power Corporation, subject "Potential Error in Generation of the Transport Correction Factors Used to Generate INCORE Analytic Factors" (KCS:93:016). The anomalies were identified during an examination of the INCORE 30 percent power maps for Robinson. In accordance with ANF-P00,002, Quality Assurance Procedure No. 3, "Design Control for Nuclear Fuel," the fuel vendor began an assessment of the potential error's impact on the Robinson cycle-16 fuel load. The assessment is scheduled to be completed by December 15, 1993. The team discussed the potential generic implications of the error with the fuel vendor. The fuel vendor indicated that the INCORE Computer Code was a Westinghouse Code and had only been used by the fuel vendor for Robinson and that Robinson had been informed of the potential error (the fuel vendor did indicate that they intended to use the code for a future reload of Shearon Harris). The team concluded that 10 CFR Part 21 requirements had been met. The team also discussed the potential for the error being applicable to other facilities which use the INCORE Computer Code. The fuel vendor stated that the error had been made by the fuel vendor personnel and was not inherent in the code. Since the fuel vendor evaluation of this issue was not complete, this item should be examined during a future inspection.

5.4.1 AIT Conclusion

The team concluded that 10 CFR Part 21 requirements had been met.

6.0 ASSESSMENT OF LICENSEE INVESTIGATION OF THESE EVENTS

6.1 Assess the effectiveness and thoroughness of the licensee's investigation of these issues.

The team reviewed their independent evaluation of the events and root causes against the licensee and the fuel vendor findings. The team concluded that the licensee and its fuel supplier have done a thorough job of review and their root cause determinations are reasonable. The licensee's and the fuel vendor's level of management involvement in their investigations and in their internal critiques of their investigations has been indepth and involved the highest levels of their respective organizations. The AIT findings basically agree with that of the licensee as noted below and in specific places in the report.

6.1.1 AIT Review of the licensee Nuclear Instrumentation Miscalibration Review Team Assessment

The licensee's investigation attributed the root cause for the nuclear instrumentation miscalibration event to be the inadequate implementation of corrective action following similar industry events. A casual factor for the event was determined to be an improper methodology used in calculating the power range currents.

Their investigation found that the licensee calculated a correction factor to apply to the previous cycle's 100 percent Power Range Nuclear Instrumentation currents by multiplying the previous current by a ratio of previous cycle average relative power for three fuel bundles to predicted average relative power for fuel bundles in the new core load. In this calculation, the two nearest, outer diagonal fuel assemblies and a third inner assembly (second diagonal) were used for relative power comparisons. This methodology differed from that recommended by the Nuclear Steam Supply System vendor Westinghouse (see figure Z).

This licensee assessment was issued on December 3, 1993. The members and their scope are outlined in Appendix C.

The AIT agreed with these findings and conclusions and in addition, found that the Nuclear Instrumentation miscalibration was the result of an incomplete understanding of the core geometry considerations by the procedure writer and inadequate review by the corporate fuels group.

6.1.2. AIT Review of Licensee's Robinson Fuel Loading Investigation Team Assessment

The licensee team considered the fuel vendor errors and their not being detected by the licensee as Principal Causes.

The licensee's investigation attributed contributing causes for the failure of a fuel inspection tool and consequent loose parts event to be:

a. inadequate tool design

b. inadequately defined roles and responsibilities

c. failure to follow proper foreign material exclusion practices.

The AIT agreed with the above findings and conclusions with the addition of the contribution of the poor physical condition of contractor and licensee personnel.

The licensee's investigation attributed the contributing causes of the misconstructed fuel bundles to be:

a. the licensee's failure to ensure fabricated fuel meets design requirements because of a lack of management direction and inadequacies in review and evaluation programs; and

b. the fuel supplier's fabrication of bundles with incorrect Gadolinium rod placement, caused by inadequacies in procedures, accountability, training and overchecks.

The AIT agreed with the above findings and conclusions.

The licensee's investigation attributed the contributing causes of the core design data problem to be:

a. Fuel supplier errors in producing the input due to the work being hurriedly done, inadequate procedures and inadequate data checkout tools.

b. Inadequate licensee oversight and review of the supplier analyses.

The AIT agreed with the above findings and conclusions.

This assessment was issued on December 8, 1993. The membership and their scope are outlined in Appendix C.

7.0 EXIT MEETING.

On December 6, 1993, the team, accompanied by the Deputy Regional Administrator for Region II, conducted a public exit meeting at the Robinson site. The licensee and NRC personnel attending this meeting are listed in Appendix D. Proprietary material has not been included in this inspection report. During the exit, the team summarized the scope and findings of the inspection. There were no dissenting comments from the licensee of the findings.

APPENDIX A

H. B. ROBINSON AUGMENTED INSPECTION TEAM (AIT) CHARTER

A. Basis

On November 16, 1993, during startup of H.B.Robinson Unit 2, reactor core flux anomalies were identified during flux mapping at approximately 30 percent power. A second flux map confirmed core design problems. Other problems were identified during the startup including the Power Range Nuclear Instruments reading 10 percent below the actual power level of 30 percent.

B. Scope

1. Develop and validate the sequence of events associated with the November 12, 1993, startup until Hot Shutdown was reached on November 17, 1993.

2. Assess the root cause and safety significance of the core neutron flux anomalies with regard to fuel and technical specification limits.

3. Determine the root cause of the miscalibrated nuclear instruments identified during startup.

4. Assess operator performance relative to the nuclear instrumentation miscalibration problem.

5. Assess the adequacy of station nuclear instrumentation calibration and refueling procedures.

6. Determine the root cause of the broken fuel handling tool event, and the effectiveness of licensee oversight of contractor fuel handling activities.

7. Assess the effectiveness and thoroughness of the licensee's investigation of these issues.

8. Assess the cause and extent of the fuel manufacturing errors at Siemens Fuel Manufacturing Facility and the extent and effectiveness of fuel verification at the site.

9. Assess the adequacy of the licensee's oversight of Siemens' fuel analysis and Quality Assurance programs.

10. Prepare a special inspection report documenting the results of the above activities within 30 days of the inspection completion.

C. Team Members

Team members will include: Team Leader, Senior Resident Inspector, Robinson Resident Inspector, Reactor Physics Specialist, License Examiner, and Vendor/Quality Assurance Inspectors to inspect at the Robinson Site; follow-up at the Siemens Fuel Manufacturing Facility will be conducted by the Reactor Physics Specialist and the Vendor/Quality Assurance inspectors.

SUPPLEMENT TO AUGMENTED INSPECTION TEAM (AIT) CHARTER FOR H. B. ROBINSON AND SIEMENS FUEL MANUFACTURING FACILITY

A. Basis

On November 16, 1993, during startup of H. B. Robinson Unit 2, reactor core flux anomalies were identified during flux mapping at approximately 30 percent power. A second flux map confirmed core design problems. Other problems were identified which included the Power Range Nuclear Instruments reading 10 percent below the actual power level of 30 percent and a broken fuel handling tool.

B. Scope

1. Determine the root cause of the broken fuel handling tool event.

2. Assess the adequacy of Siemens' Quality Assurance program for the manufacture of special fuel tools.

3. Assess the cause and extent of the fuel manufacturing errors at Siemens Fuel Manufacturing Facility and the extent and effectiveness of fuel assembly verification at Siemens.

4. Assess the adequacy of the licensee's oversight of Siemens' fuel analysis and Siemens' Quality Assurance programs.

5. Assess the Siemens analysis of the core neutron flux anomalies.

6. Assess the effectiveness of Siemens' program for notifying licensees of known deficiencies in either hardware or services provided.

7. Prior to exiting the Siemens' facility brief the AIT team leader of the preliminary inspection findings via telephone.

8. Provide inspection results in writing to AIT team leader within one week of exiting the Siemens' facility.

C. Team Members

Team members will include: Reactor Physics Specialist - Edward D. Kendrick and Vendor/Quality Assurance Inspectors - Steven Matthews and W. H. Rogers.

APPENDIX B

AIT REVIEW TEAM

Corrective Action Assessment Team

Warren Dorman	Team Leader, RNP CAP/OEF
Franklin Murray	HPES RNP
	BND
	HNP

SCOPE: Review past RBN performance (NAD, INPO, NRC, and other assessments) and evaluate effectiveness of RBN CAP program in correcting previously identified performance issues and predicting areas requiring additional attention.

Robinson Fuel Loading Investigation Team

Lou Martin	Team Leader
Bob Toth	INPO, Assistant Team Leader
Dave Waters	(Misconfiguration Focus)
John Eads	(Inspection Tool Failure Focus)
Jim Thompson	(Power Escalation Recommendation Focus)
Outside Member	(Assist With Fuel Fabrication Focus)

SCOPE:

(1) Conduct a detailed root cause analysis of the core loading problems of the following:

- The Siemens inspection tool failure and the resulting fuel assembly relocation.
- The misconfiguration of fuel assemblies.
- The adequacy of the licensee's oversight of Siemens activities both onsite refueling and the core analysis activities.

(2) Review documentation and interview personnel at HBR, Fuels, and Siemens to determine root cause and why not identified by the licensee and Siemens prior to fuel load.

(3) Evaluate casual factors for other impact.

(4) Complete tasks prior to head replacement.

Nuclear Fuels Instrumentation Investigation Team

C. S. Hinnant	Team Leader
Chip Moon	Operations
Bryan Waldsmith	Operations
Danny LaBelle	Fuels
Jo Ellen Westmoreland	Reactor Engineer
Franklin Murray	HPES
Dick Cady	CAP/Maintenance
Brian O'Donnell	INPO
David Coates	Training

SCOPE: Conduct investigations following the following guidelines:

(1) Operations: time line; events and casual factor; ERFIS data; Operations logs; plant data; and power ascension coordination

(2) Fuels: fuel vendor data; corporate fuel design data; fuels/plant interface.

(3) Reactor engineering interface with fuels and operations and comparison of Harris lessons learned with HBR corrective actions.

(4) HPES: Event analysis using "yellow sticky" method and HBR actions from similar O.E. identified events.

(5) CAP/Maintenance: PL-026 and format of/process to develop final report.

(6) INPO: Industry experiences.

(7) Training: Reactivity management training, and startup from RFO training.

APPENDIX C

ATTENDANCE LIST AT EXIT DECEMBER 6, 1993

S. A. Bilings	Regulatory Affairs
T. A. Peebles	AIT Leader, RII
L. A. Reyes	Deputy Regional Administrator, RII
E. D. Kendrick	Nuclear Engineer, NRR
C. R. Ogle	AIT Member, RI
B. H. Rogers	Reactor Engineer/UIB, NRR
S. M. Matthews	Quality Assurance Engineer DRIL/VIB, NRR
D. Waters	Manager, Regulatory Affairs, CPO
S. Zimmerman	Manager, Nuclear Fuel Management & Safety Analysis
M. Pearson	Plant General Manager, Robinson
C. R. Dietz	VP, RNPD, CP&L
H. W. Habermeyer, Jr.	VP, NSD, CP&L
W. S. Orser	Exec VP, Nuclear Generation
R. E. Rogan	Manager, CP&L, Licensing
B. H. Clark	Manager, Maintenance
A. R. Wallace	Manager, Licensing/Regulatory Programs
M. Herrell	Manager, Training
T. P. Cleary	Manager, Technical Support
J. Guibert	Consultant to CP&L
D. G. McAlees	Sr. VP & GM, Siemens, Nuclear Division
[illegible]. N. Morgan	VP Engineering, SPC-ND
[illegible] Watts	Electrical Dept-SCPSC
R. S. Stancil	Nuclear Bus. Oper., CP&L
S. Singh Bajwa	NRC, NRR, Projects
B. L. Mozafari	NRC, NRR, PDII-1
P. J. Jordan	Manager, Nuclear Human Resources, CP&L
W. S. Baum	Nuclear Employee Relation, CP&L
G. Newsome	Nuclear Engineer, CP&L
W. Pridgen	CP&L, Manager
C. S. Olexik	Manager, Plant Assessment, CP&L
K. Clark	Public Affairs, NRC, RII

H.B. ROBINSON CORE

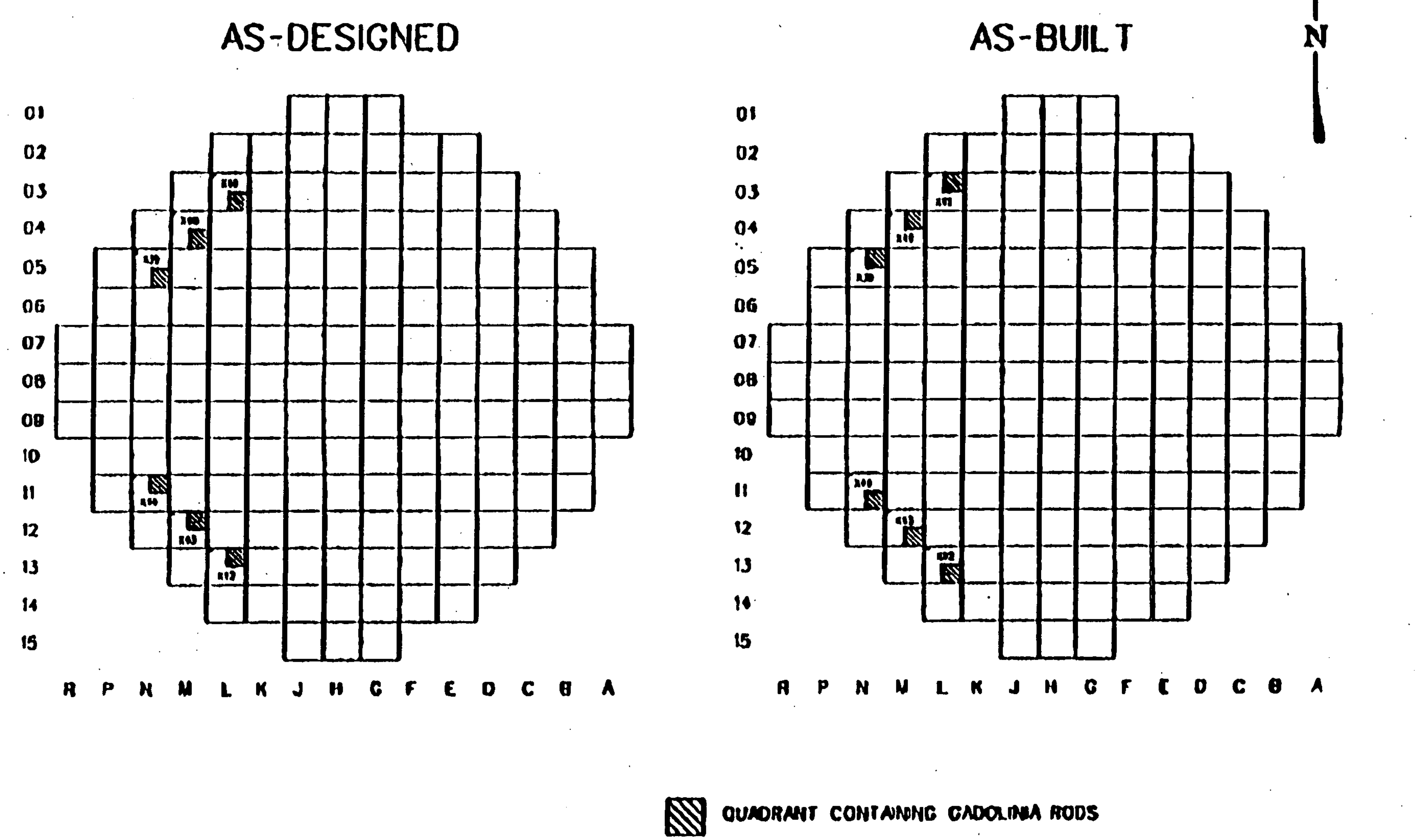


FIGURE X
FUEL INSPECTION TOOL

FIGURE Y

GADOLINIUM FUEL ASSEMBLY AS DESIGNED vs AS-BUILT

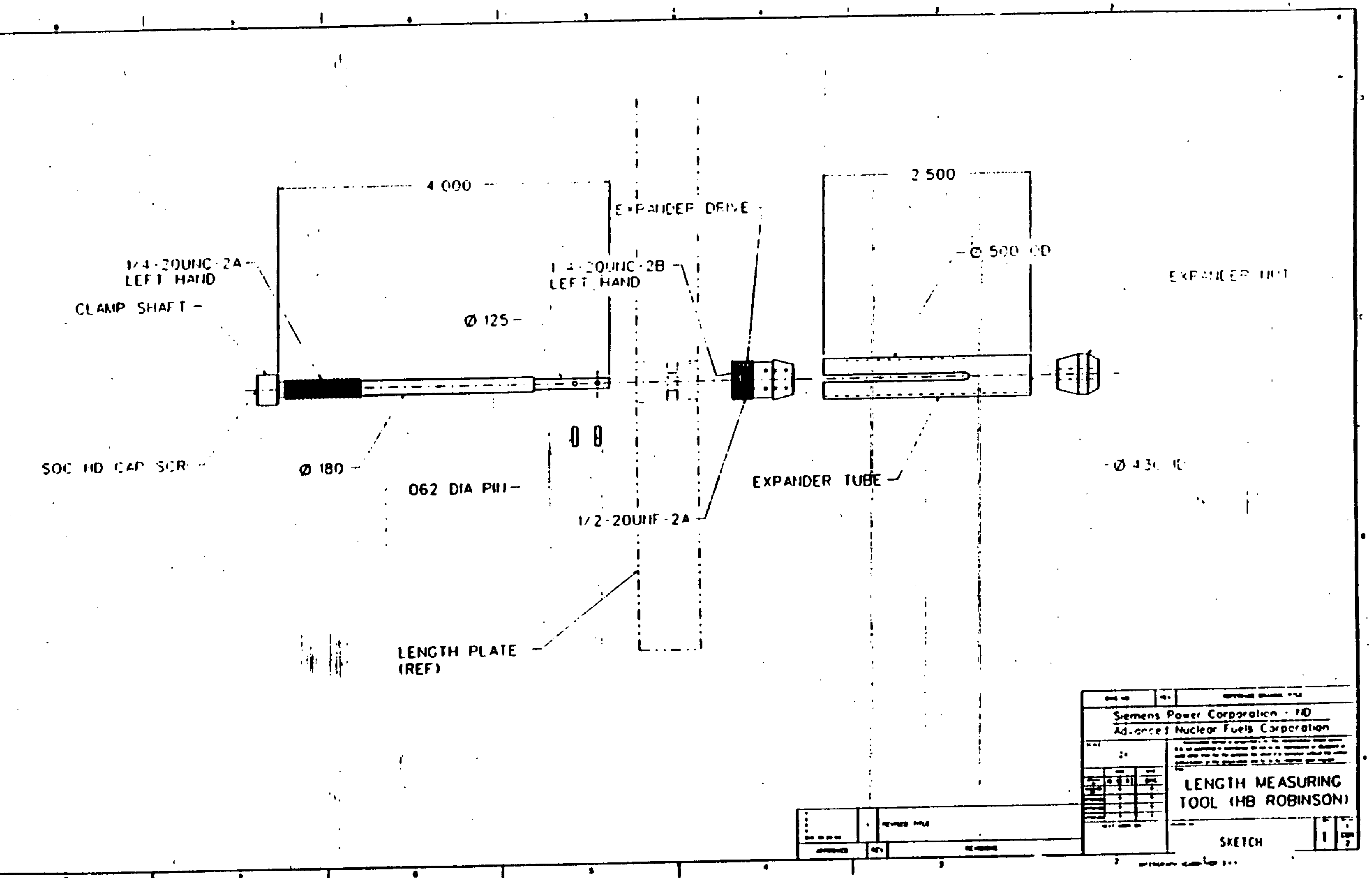
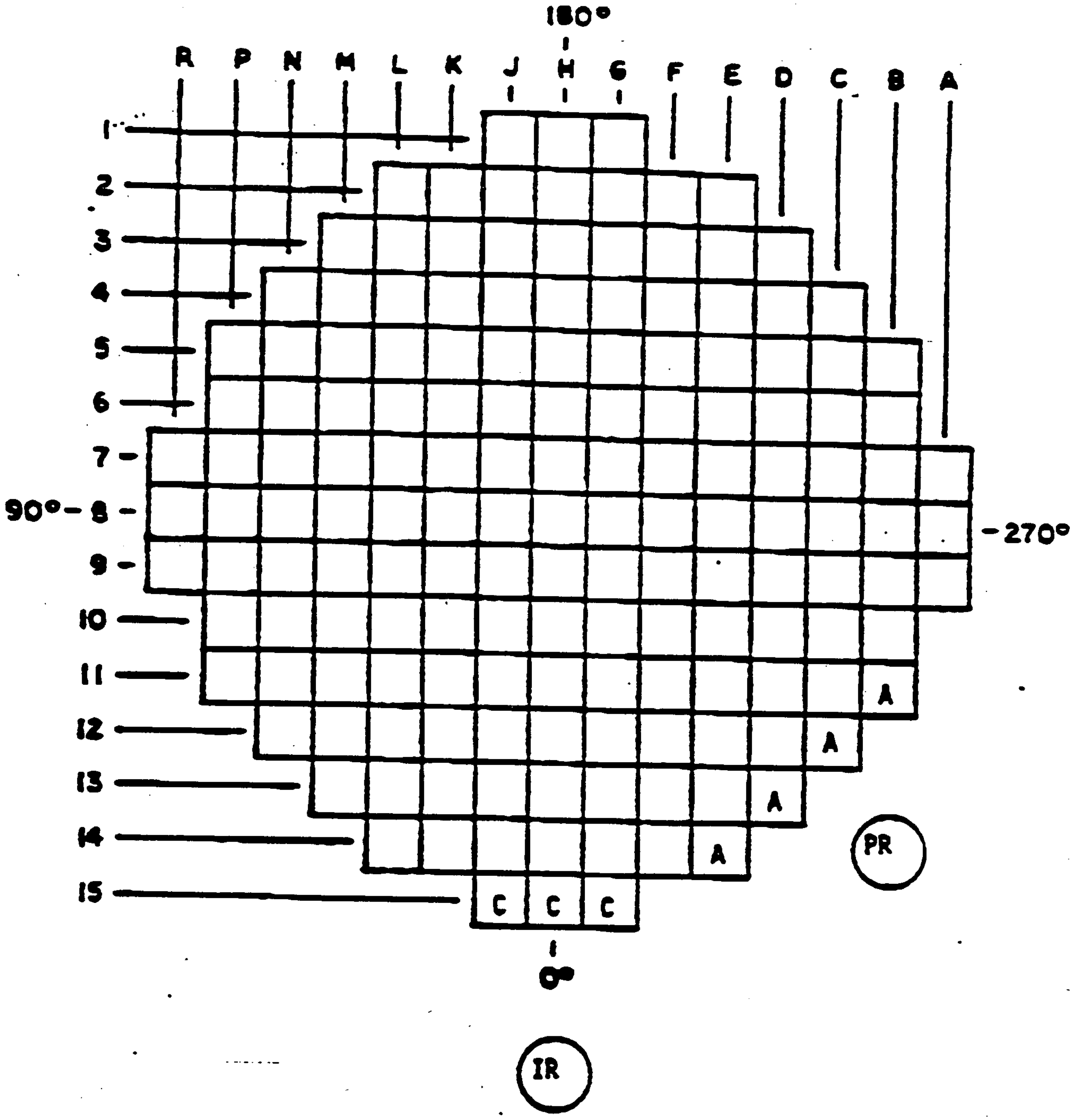


FIGURE Z

RECOMMENDED ASSEMBLY TO USE FOR NI CALIBRATION



Three Loop Plant