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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 10, 1993

The Honorable Donald W. Riegle, Jr.
United States Senate
Western Regional Office
Suite 716 Federal Building
110 Michigan Avenue, N. W.
Grand Rapids, MI 49503

Dear Senator Riegle:

This is in response to your July 21, 1993, letter to A. Bert Davis inquiring about a fuel rod breakage identified on July 1, 1993, at Consumers Power Company's Palisades Nuclear Power Plant.

You requested a copy of our investigation report into this matter after it had been completed. An NRC Augmented Inspection Team (AIT) conducted a special inspection at the site on July 8 through 20, 1993. This special onsite review is documented in the attached inspection report.

All our inspection reports are public documents; they are routinely sent to local Public Document Rooms near the subject facilities. In addition, in the case of this AIT inspection, a public meeting was held at the conclusion of the inspection on August 20, 1993. At that time, several members of the public, including your constituents, also requested copies of the AIT report. We are fulfilling each of those requests.

I trust this information is responsive to your needs.

Sincerely,

James M. Taylor
Executive Director
for Operations

Enclosure:
AIT Inspection Report

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137-5927

August 31, 1993

Docket No. 50-255

Consumers Power Company
ATTN: Mr. Gerald B. Slade
General Manager
Palisades Nuclear Generating Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

SUBJECT: NRC REGION III AUGMENTED INSPECTION TEAM (AIT) REVIEW OF THE JULY 1 AND JULY 6, 1993, PALISADES FUEL HANDLING EVENTS

Dear Mr. Slade:

The enclosed report refers to a special onsite review conducted by an NRC Augmented Inspection Team (AIT) from July 8 through July 20, 1993, relative to the broken fuel rod discovered in the reactor cavity tilt pit on July 1, 1993, and the lifting of the reactor upper guide structure with a fuel assembly stuck underneath on July 6, 1993, at the Palisades Nuclear Plant. The team leader was Mr. William Dean of the Office of Nuclear Reactor Regulation (NRR) and the team was composed of Messrs. Robert Lerch, Andrew Dunlop, John House, Kombiz Salehi and Charles F. Gill of this office and Messrs. James Davis, Anthony Hsia, and Shih-Liang Wu of NRR. The report also refers to the followup activities of your staff. At the conclusion of the inspection, a public management meeting was held on July 20, 1993, with David Hoffman, Vice President, Nuclear Operations, you, and members of your staff to discuss the inspection findings.

Areas examined during the inspection are identified in the enclosed AIT report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

The AIT was formed to gather information related to the events. The team examined your response to the events, your efforts to determine the root cause and failure mechanisms, recovery plans and procedures, the effectiveness of the quality assurance organization, performance related issues, fuel failure identification activities, and corrective actions. Enforcement actions resulting from the issues identified will be determined separately from this inspection.

Any events such as the broken fuel rod and the stuck fuel assembly are viewed as serious, even though the consequences posed no threat to public health and safety. Your response to the events required carefully considered actions to minimize the risk of excessive radiation exposure to plant personnel and to safely recover the broken fuel rod and the stuck fuel assembly. Recovery actions were satisfactorily accomplished, though there were some initial performance related problems in freeing the stuck fuel assembly. The AIT noted that no operational safety parameters were approached or exceeded.

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Notwithstanding, the AIT concluded that management expectations were not consistently translated to the working levels as indicated by the repeated work performance issues identified by the team. The team also determined that there was a non-conservative approach in some of the decisions regarding plant operations, particularly the insufficient evaluation of I-series fuel assemblies prior to installing them in the core for a sixth cycle of operation. The team also concluded that a questioning attitude was not prevalent within all levels of the organization. This is exemplified by the less than thorough evaluation of radiochemistry data which indicated failure of a low power fuel rod. In addition, the team found that NPAD was ineffective in countering those problems and weak in identifying problems in operations and assuring that corrective actions are adequate. During the inspection, the team observed a number of onsite activities, and noted that increased management attention resulted in improved work performance. The team also noted that a thorough and thoughtful root cause analysis of each event was being pursued.

Our assessment was that your efforts were thorough and well-developed however, your root cause analyses, and corrective actions were not yet completed. As a result, the team evaluated those that were completed. We understand that, as documented in the Confirmatory Action Letter (CAL) dated July 8, 1993, you will discuss the final results of your root cause analyses and corrective actions with senior NRC management prior to the resumption of power operations. We will continue to closely follow your efforts to determine root causes and implement corrective actions.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

William L. Forney
Acting Director
Division of Reactor Safety

Enclosure: AIT Inspection
Report No. 50-255/93018(DRS)

See Attached Distribution

Distribution

cc w/enclosure:
David P. Hoffman, Vice President
Nuclear Operations
David W. Rogers, Safety
and Licensing Director
OC/LFDCB
Resident Inspector, RIII
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AUGMENTED INSPECTION TEAM REPORT

U.S.NUCLEAR REGULATORY COMMISSION

PALISADES BROKEN FUEL ROD AND STUCK FUEL ASSEMBLY

INSPECTION REPORT NO. 50-255/93018(DRS)

JULY 8, 1993 TO JULY 20, 1993

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/93018(DRS)

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
1945 West Parnall Road
Jackson, MI 49201

Facility Name: Palisades Nuclear Plant

Inspection At: Palisades Site, Covert, MI

Inspection Conducted: July 8 - 20, 1993

Inspectors: Robert M. Lerch, DRS
Andrew Dunlop, DRS
John House, DRSS
Charles F. Gill, DRS
James A. Davis, NRR
Anthony H. Hsia, NRR
Shih Liang Wu, NRR
Kombiz Salehi, DRS

Approved By: Robert M. Lerch for
William M. Dean, NRR
Team Leader
8-31-93
Date

Edward G. Greenman, Director
Division of Reactor Projects
8-31-93
Date

Inspection Summary

Inspection on July 8 - 20, 1993 (Report No. 50-255/93018(DRS))

Areas Inspected: Augmented Inspection Team (AIT) inspection conducted in response to the broken fuel rod discovered on July 1, 1993, and the stuck fuel assembly event of July 6, 1993, at the Palisades Nuclear Plant. The review included validation of the sequence of events, evaluation of the licensee's failure mechanism determination and root cause analyses, review of the effectiveness of previous corrective actions associated with stuck assembly events, assessment of licensee's evaluation of potential precursors to the fuel rod failure, evaluation of ongoing refueling activities, assessment of the effectiveness of management involvement and the quality assurance organization related to these events, and evaluation of the licensee's corrective actions.

Results: A summary of the AIT results are contained in Section 2.2; Broken Fuel Rod Assessment, Section 2.4; Stuck Fuel Assembly Assessment Summary; and Section 2.5, Safety Summary.

1.0 Introduction

1.1 Scope of Inspection

On July 1, 1993, at the Palisades Nuclear Plant, a broken fuel rod was identified in the tilt pit area of the reactor cavity. On July 6, 1993, while removing the upper guide structure (UGS) as part of the investigative activities associated with the broken fuel rod, a fuel assembly was inadvertently lifted from the core. In response to these events, the Regional Administrator sent an Augmented Inspection Team (AIT) to the site to document and validate the relevant facts, determine the probable causes, and evaluate the licensee's analyses efforts and review of the events including corrective actions. Also, the team was to determine the adequacy of management involvement, effectiveness of previous corrective actions, effectiveness of the quality assurance organization, and responsibility for core design. The charter of the AIT (Attachment A) was developed and approved on July 8, 1993, concurrent with the issuance of a Confirmatory Action Letter (Attachment B), which included several items to be accomplished under the cognizance of the AIT.

The NRC AIT held an entrance meeting with plant management and staff on July 8, 1993, and performed the inspection during the period of July 8-20, 1993. A public management meeting was held with plant management on July 20, 1993. Attachment C lists the attendees at the entrance meeting and Attachment D lists the individuals who attended the exit meeting.

1.2 Team Composition

The AIT consisted of a team leader from the Office of Nuclear Reactor Regulation (NRR), the headquarters project manager, two headquarters specialists, and five regional specialists. The team's combined expertise included refueling operations and procedures, materials, reactor vessel internals and core/fuel assembly design, radiochemistry, health physics, plant operations and management controls.

2.0 Executive Summary

2.1 Broken Fuel Rod Event

While draining the reactor cavity tilt pit on June 30, 1993, as the plant was emerging from refueling activities, elevated radiation levels were noted. Visual inspection of the tilt pit area noted a rod-like object that was confirmed to be part of a broken fuel rod from fuel assembly I-24. Two other segments of the rod were identified, making up all but about one foot of an entire fuel rod. The water level in the tilt pit was raised to minimize the radiation hazard to plant personnel. The rod segments were recovered and placed in two storage baskets on July 6. It was noted that one segment, approximately 5 feet in length, had an axial split with none of its fuel pellets present. It also appeared that a section(s) of cladding material was missing from the entire length of this segment. The other two pieces were intact and appeared to contain fuel pellets. The remaining top section of fuel rod was found still contained within the I-24 assembly when it was removed from the core on July 13.

While removing the upper guide structure (UGS) on July 6, a fuel assembly (SAN-8 in core location Z-11) was found stuck to the UGS as it was being lifted. This event is summarized in Section 2.3. This resulted in cessation of refueling activities until the licensee could conduct a prompt assessment of these two events and determine appropriate near-term corrective actions. The results of this assessment were provided on July 12 to an Augmented Inspection Team (AIT) which had convened on July 8. Team concurrence was given to resume refueling operations and the damaged fuel assembly was removed from the core and transferred to the spent fuel pool on July 13.

2.2 Broken Fuel Rod Assessment

The team concluded that though the fuel failure could not reasonably have been predicted, the licensee acted in a less than conservative manner in several instances which contributed to creating the situation. The I-24 fuel assembly was one of sixteen I-series assemblies that were being placed in the core for a sixth cycle of operation. The licensee had no other experience with assemblies that had been in the core for five, much less six, cycles of operation. Additionally, the fuel vendor, Siemens Power Company (SPC), had little experience with pressurized water reactor (PWR) fuel that had been in the core for so many cycles. However, no inspections other than a visual examination of one of the bundles were performed during the outage, and no tests were conducted to assess the integrity of any of the fuel rods.

During the operating cycle prior to the outage (cycle 10), radiochemistry data indicated a steadily increasing trend of fission product activity, though classical iodine spiking was not observed. The absence of some of the typical indicators of failed fuel caused the licensee's technical staff to conclude that the elevated fission products were due to "tramp uranium" from fuel leaks that occurred during the previous cycle (cycle 9). Other than the fuel vendor, who confirmed the licensee's supposition, no independent or outside assistance was formally sought to validate their opinion. Opinions from Westinghouse and a corporate health physicist supporting a fuel failure were given limited credibility. The possibility of a fuel rod failure was discussed, and was discounted. The team believed that the licensee's assessment, though possible, was not the most plausible explanation for the observed indications. The licensee's procedure for detecting fuel failures was ineffective in detecting leaks in low power, peripheral assemblies. The licensee also did not treat the possibility that a peripheral fuel rod may be leaking with a conservative approach. They did not alter their limited inspection plans for the I-series assemblies even though the potential that a fuel leak in a peripheral rod may have occurred.

2.3 Stuck Fuel Assembly Event

On July 6, 1993, while the upper guide structure (UGS) was being lifted from the reactor vessel, fuel assembly SAN-8 was observed stuck to the bottom. The UGS had been lifted three feet and, per procedure, a surveillance camera was used to detect any stuck fuel assemblies. The stuck assembly was discovered and an Unusual Event was declared. On July 7, unsuccessful attempts were made to disengage the SAN-8 assembly from the UGS by alternately adjusting the tension on two chainfalls which were attached to the assembly by "j-hooks".

The licensee also made an unsuccessful attempt to free the assembly with the use of a slide hammer which was too heavy to effectively manipulate. On July 8, using an improved slide hammer, the licensee applied a striking force several times to the fuel assembly upper tie plate. This freed the assembly from the UGS. The assembly was safely lowered back into the core and the UGS was then placed on its storage pads and the Unusual Event terminated.

2.4 Stuck Fuel Assembly Assessment

The team concluded that the lift methods for the UGS may have been a contributing factor but could not be directly attributed to causing the stuck assembly. The root cause analyses performed by the licensee following the 1988 and 1992 stuck assembly incidents were extensive but indeterminate and ultimately unsuccessful in preventing a recurrence of the event. Corrective actions from these previous events resulted in increased supervision of UGS lift activities and early identification of the stuck assembly. The approach for the 1993 root cause determination was well considered and used multi-disciplined teams including a consultant.

During the inspection, the team noted that management was not effective in several areas such as job briefings, procedural adherence and industrial safety. Review of the working copy procedures and interviews with personnel involved in the UGS lift determined that procedures specifying the use of load cell equipment were not followed. During recovery of the stuck assembly, the team observed job briefing weaknesses; inadequate communication by the licensee supervisor with the contractor supervising the job; and workers exceeding procedural limits for chainfall tension, climbing on railings without safety harnesses, and removing safety glasses.

2.5 Safety Assessment Summary

Neither of the events posed a threat to the health and safety of the public. Both events occurred inside the containment building and resulted in no measurable radiological release to the environment. With respect to the potential of dropping an irradiated fuel assembly onto the core, this event was bounded by the analysis contained in the Final Safety Analysis Report (FSAR). Also, improvements to operating procedures that were made after the stuck assembly event in 1992 resulted in early detection of the stuck assembly, greatly minimizing the potential effect of a dropped assembly.

Both of these events however, have potential implications for the safety of plant personnel. The broken fuel rod resulted in elevated radiation readings in the area of the tilt pit, which if not discovered, could have resulted in a higher dose to plant personnel performing activities in that area of the reactor cavity. Fuel that is not recovered and which disperses throughout the primary coolant system (PCS) may complicate future operations by spreading hot particles to unwanted areas of the PCS, increasing general radiation levels, and making it more difficult to detect fuel failures. Unrecovered fuel cladding may create a loose parts hazard.

During activities associated with the stuck fuel assembly, some workers did not comply with good industrial safety practices.

With respect to meeting management expectations, the licensee did not consistently carry out refueling activities in an orderly and controlled manner. Some of the problems identified by the team, particularly those related to contractor performance, were similar to problems both the licensee and NRC had identified during the last refueling outage. Also, similar problems were identified recently by Region III inspectors investigating the event when the reactor vessel head was lifted with a control rod still attached on June 15, 1993.

During the inspection, the team noted a significant increase in the level and effectiveness of management involvement. Efforts to determine the failure mechanism, root cause, and appropriate course of action to recover the stuck assembly and reconstitute the core were conducted with a measured and thorough approach. Due to the complex nature of the fuel failure and the number of potential causes considered by the licensee, these efforts were still in progress at the end of the inspection.

3.0 Broken Fuel Rod Event

3.1 Description of the Event

On June 28, 1993, with the plant in the refueling mode, the licensee was draining the reactor cavity as part of post-refueling activities. Health physics personnel noticed radiation levels increasing during this activity. Subsequent surveys found a 900 R/hr hot particle in the reactor cavity. The crew removed the particle and placed it in a lead container. Radiation protection management investigated the incident and requested their staff to examine applicable plant systems, tanks, and drains for the possible presence of similar particles. At 1045 on June 29, 1993, the workers identified high activity in the containment drain piping at the 590 foot elevation. No immediate actions were taken to analyze the high activity sources (See Section 3.3). (Note: On July 17, analysis of the hot spot revealed no fission products in the containment drain piping.)

Refueling operations continued and as the water level in the tilt pit area was lowered, the dose rate on the refueling bridge increased to 700 mrem/hr. This prompted a survey of the tilt pit and at 2330 on June 30, a rod shaped item was found that had dose rates of 5,000 to 8,000 R/hr on contact. By July 1, further investigation confirmed the presence of three segments which were classified as being part of a fuel rod. The combined length of the three segments constituted about 11 feet of the total 12 foot length of a fuel rod, with the upper segment still missing. On July 2, the fuel rod was positively identified which allowed the licensee to associate the broken fuel rod with fuel assembly I-24. The broken rod was noted to be from a corner of the assembly which had been located in the corner of the core adjacent to the core shroud during the previous operating cycle (cycle 10). Assembly I-24 was still in the reactor for cycle 11.

A section of the broken fuel rod about 5 feet in length was split axially and was missing a shard of cladding material that appeared to be of the same length. Also, there were no fuel pellets observed within or around this segment. The other two segments, which were about 4.5 and 1.5 feet in length, appeared to be intact and fully loaded with fuel pellets. The three visible pieces of the broken fuel rod were recovered and placed in two specially constructed storage baskets on July 6. The two baskets were later placed in a storage can and transferred to the spent fuel pool.

On July 6, while disassembling the reactor to inspect assembly I-24, the licensee inadvertently lifted another fuel assembly (SAN-8) that was stuck to the UGS. This event is addressed in detail in Section 4.0. All refueling activities other than those associated with recovery of the stuck assembly were halted. The NRC augmented inspection team (AIT) was formed and arrived onsite on July 8. After reviewing the results of the licensee's prompt assessment and near-term corrective actions, the team concurred in the decision by the licensee to resume refueling activities. The I-24 bundle was removed from the reactor vessel at 0745 on July 13 and transferred to the tilt pit area. The missing upper fuel rod segment was found within the upper grid spacer during a video camera inspection prior to moving the assembly. However, this piece of fuel rod fell out during transfer of the assembly to the tilt pit. On July 13 at 1345, the I-24 assembly was transferred to the spent fuel pool. The last segment of the broken rod was retrieved from the top of fuel assembly N-62 (see Figure 1) on July 13 at 2215. It was placed in the second storage basket noted above.

3.2 Description of Equipment

Fuel Assembly: A typical fuel assembly consists of an upper tie-plate, 10 grid spacers, a lower tie-plate, and a 15 by 15 array of fuel rods. The distance from the bottom of the upper tie plate to the top of the lower tie plate is 140.7 inches (See Figure 2). The spacers (See Figure 3) are generally distributed evenly along the length of the fuel assembly 15.5 inches apart, although the bottom two spacers and the upper two spacers are slightly closer together. The spacers are spot welded to eight Zircaloy-4 guide bars that are attached to the upper and lower tie plates. There are two guide bars in peripheral slots on each face of the spacer fuel assembly. Fuel rods are securely held in the spacers using Inconel 718 springs. The Zircaloy-4 spacers with inconel springs are called bi-metallic spacers. There are eight guide tubes in the interior of the fuel assembly that extend from the upper tie plate to the lower tie plate. The guide tubes are 0.416 inches in diameter. A cluster plate has been placed in the I-series assemblies that hold eight full length hafnium rods that are inserted into the eight guide tubes. The hafnium rods are used to reduce thermal neutron fluence as part of the licensee's reactor vessel beltline flux reduction program. Finally, there is a full length instrument tube in the center of the bundle.

Fuel Rod: A fuel rod (See Figure 4) consists of a stack of approximately 470 UO_2 pellets with alumina discs at each end and a compression spring at the top end clad within Zircaloy-4 tubing and sealed by welding end caps to each end.

The rod is pressurized with helium and a plenum is provided at the top of the fuel column to accommodate the gaseous fission products released from the fuel and to absorb axial expansion of the fuel. The total length of a fuel rod is approximately 12 feet.

Refueling Machine: The refueling machine (RM) is a hoist on a traveling bridge and trolley which spans the reactor cavity and moves on rails located on the working floor of the containment area. The RM is used to lift and move fuel assemblies under water from the tilt machine to the reactor and back. The hoist assembly contains a coupling device, which when rotated by the actuator mechanism, engages the fuel bundle or control rod to be removed. The hoist assembly is moved in a vertical direction by a cable attached to a hoist winch. Once the fuel bundle is raised into the hoist assembly, the hoist assembly is raised into the refueling machine mast, and the refueling machine can then be moved throughout the reactor cavity area.

During withdrawal or insertion of a fuel assembly, the load on the hoist cable is monitored at the control console to ensure movement is not restricted. Variation in excess of 10% of normal load will automatically stop the hoist. A zoned mechanical interlock is provided which prevents the opening of the fuel grapple and protects against inadvertent dropping of the fuel. A spreader device is provided which spreads adjacent fuel bundles to provide unrestricted removal and insertion.

Tilt Machine: The tilt machine in containment (there is another one in the spent fuel pool) consists of a fabricated hollow rectangular structure, supported through a pivot to a triangular-shaped support base. This structure is closed at one end and open at the other, allowing the transfer carriage to move completely into the structure. Hydraulic cylinders attached to the box and the frame are provided to rotate the transfer carriage to a vertical or a horizontal position. Interlocks are provided to ensure safe operation of this equipment by prohibiting lowering of a fuel assembly unless the transfer carriage is properly positioned in the tilt machine and preventing inadvertent rotation of the tilt mechanism while a fuel assembly is being lowered.

3.3 Fuel Fragment Characterization and Assessment

On June 28, 1993, during decontamination of the reactor cavity at the end of the cycle 10 refueling outage, a fragment was found adjacent to the reactor vessel. The fragment was about the size of a pencil eraser and had a dose rate of 900 R/hour on contact. Since the reactor cavity is decontaminated at the end of each refueling outage, it was assumed that the fragment was deposited during the cycle a refueling outage. Radiochemical analyses performed (gamma spectroscopy) of the fragment on June 29 showed the presence of fission products (isotopes) indicating that the fragment was fuel. The licensee initially stated that they believed the fuel fragment came from known fuel failures during cycle 9 operations. However, as the investigation of the broken fuel rod progressed and additional information became available, including the estimated loss of 219 fuel pellets, the licensee stated that the cycle 10 rod failure was the likely source of the fragment. At the end of the inspection, the licensee was attempting to determine if the fuel fragment was from cycle 10 based on computer analyses of burnup, neutron flux and isotopic decay. Using fuel parameters for the failed fuel rod from the I-24 bundle, the licensee believed that the fuel fragment was from that rod. In retrospect, the radiochemistry data that existed at the time the plant was shutdown indicated a failure of a low power, peripheral fuel rod. The team concluded that there has been no other event or information that would support the existence of another significant fuel failure.

3.4 Fuel Rod Failure Mechanism

Fuel rod AT300057 from fuel assembly I-24 broke into 4 pieces plus a shard(s) of cladding 4-5 feet long. Extensive video camera examinations were performed to ascertain the damage to the fuel rod and assembly I-24 (See Figure 6). The top piece was about 7 inches long and included the top end cap. The top piece was located in its original location in spacer 10 in fuel assembly I-24. The fracture between the top piece and the second piece appeared to be a green stick failure. A green stick failure is characterized by significant crack growth and plastic deformation prior to fracture. The second piece was about 5 feet long and did not contain any fuel pellets but did contain the plenum spring. All of the spacer corners retaining the second section of fuel rod were missing. The fracture appearance between the second and third piece also appeared to be a green stick fracture. The spacers originally retaining the third piece were fractured at the corner weld but otherwise intact with the exception of spacer 3 that was missing a piece of the spacer corner. Piece 3 contained fuel pellets. The fracture appearance between piece 3 and 4 was a flat fracture with no evidence of a green stick fracture. Piece 4 is about 1½ feet long and includes the lower end cap. Two grid spacers held this piece in place. The upper spacer was split at the weld. The bottom spacer was torn at the top of the corner but intact at the bottom.

A torn grid spacer is usually indicative of coolant flow induced fuel rod vibration. To cause excessive flow vibration on fuel rods requires complete or partial relaxation of the Inconel springs of a grid spacer. From observing the fuel inspection video, the team noticed that there was slight fuel rod movement at other corners which confirmed the presence of spring relaxation. In some cases, the missing grid spacer was extended to the adjacent grid cells such that the adjacent fuel rods were exposed. There were a few missing Inconel springs which may indicate an external mechanical force interfering with the grid spacer.

Individual fuel rod examinations involve two types of tests. The first test is a pull test where a fuel rod is pulled completely out of the fuel assembly while the pull force is monitored. As the fuel rod exits each spacer, the decrease in pull force is recorded. The pull force for each spacer is related to the remaining spring force for that spacer. The second test is an eddy current test (ECT) of the fuel rods. The ECT reveals if the fuel rod contains any holes or fretting wear.

The licensee presented an inspection plan for the fuel rods to the AIT. The plan was to pull all four corner rods, four peripheral rods, and two interior rods and to conduct the ECT on these same rods. The plan was to examine fuel assembly I-24 followed by I-21 and I-48. I-21 was in a symmetric location

(X-19) (See Figure 7) to I-24 (B-19) and I-48 was in position B-19 during cycle 9. These three fuel assemblies were initially in a peripheral location in cycle 5, then placed in the center of the core for cycles 6 and 7. They were then placed in peripheral locations for cycles 9 and 10. All three of these fuel assemblies showed loose corner fuel rods during the pull test.

As the licensee continued their root cause analysis, other fuel assemblies were examined including visual, eddy current, and pull tests. Attachment E summarizes all the test results made available to the team by the end of the inspection. These inspection results essentially demonstrated that there were noticeable wear indications on fuel rods that had been in corner locations adjacent to the shroud. It appeared that there were no indications on any center rods or other peripheral rods, except those adjacent to the corner rods. This indicates that the rods were probably loose, and the indications are probably the result of spacer fretting. No through-wall indications were seen, although several were greater than 85%, particularly on the I-series assemblies that had been in the core for 5 cycles and in corner shroud locations.

At the end of the inspection, the licensee had not yet completed their analysis of the failure mechanism. That the fuel rod originally failed while in the core during operating cycle 10 was indicated by a plume of fuel material found on the upper portion of the assembly. A possible scenario was that, during operation, some of the spacers failed due to rubbing against the shroud and a fracture of the rod occurred. Then, while transferring the assembly between the core and the tilt pit, the broken segment of rod caught on the internals of the refueling machine mast or tilt machine, creating several other fractures and causing the pieces to fall to the tilt pit floor. The NRC will assess the licensee's final determination of the failure mechanism after it is completed.

3.5 Failure Indications

The licensee noted increases in fission isotopes, including Np-239, I-134, I-131, Cs-137, and gross gamma activities, in the primary coolant beginning in late 1992 (See Attachment F, Figures 1-4). In reviewing selected gamma spectroscopic data, the team noted that total fission gases (Kryptons and Xenons) had also increased along with the above mentioned isotopes. The licensee had not evaluated this data, focusing only on the behavior of Xe-133, which is a typical indicator of a fuel failure. Fission product activity in reactor coolant increased for approximately 6 months until the refueling outage began in June 1993. The fission isotope concentrations increased by factors of 5-8 during this period. The licensee noted that the overall activity levels of the reactor coolant were very low. They also noted that even though they attempted to induce iodine and xenon spiking during several power reduction evolutions, no spikes, which are often characteristic of fuel rod failures, were observed.

During cycle 9 operation, the licensee determined that at least 3 fuel rods had failed based on standard isotopic analysis. Failure of a fuel rod is usually accompanied by a significant release of fission gases and iodine, along with other fission products. Licensee representatives stated that since the cycle 10 data did not fit this classic rod failure pattern, they did not believe that there was a very high probability that a rod had failed.

The continuously increasing levels of fission isotopes were thought by the licensee to have originated from "tramp uranium" released during the known cycle 9 fuel failures. The licensee stated that the "tramp uranium" had plated out on surfaces inside the reactor coolant system including inside the steam generator (S/G) tubes. During operation, reactor coolant pH is controlled by the concentration of boric acid and lithium hydroxide in the reactor coolant. During November 1992, as the boron levels decreased, the amount of lithium hydroxide was reduced (delithiation) to maintain the required pH range. The licensee postulated that the delithiation process and the associated small pH shifts caused the plated out "tramp uranium" to go into solution and be transported back through the reactor core where the neutron flux resulted in the production of fission isotopes.

Documents dated March 18, April 19, and May 19, 1993, that were prepared by the licensee's fuel performance team, indicated that although there was some uncertainty in the data, a fuel rod failure probably did not exist. The licensee's fuel vendor, Siemens Power Company (SPC), reviewed the data in early March and concluded that there had been no fuel failure. This data was also provided to another fuel vendor on an informal basis and on April 14, 1993, this fuel vendor concluded that the radiochemistry data was consistent with a fuel failure and that similar fuel problems had been observed in a European pressurized water reactor.

The team discussed the significance and interpretation of the radiochemistry data with the licensee and noted that this data was indicative of a low power fuel rod failure. This type of rod failure had occurred in other reactors and should have been given more consideration by the licensee's technical staff, especially after the licensee was made aware of this information. The team noted that their procedure for monitoring fuel performance, RSA-03, "Fuel Performance Monitoring," used the INPO fuel failure criteria, which is based on the magnitude and not the rate of change of radiochemistry data. Also, other than the informal request to another fuel vendor, the licensee did not effectively use other industry resources such as INPO's NOTEPAD system or contact other utilities to solicit advice. The licensee also failed to use other industry methods such as those of the Electric Power Research Institute, which take into account the rate of isotopic activity increase. Instead, the licensee took a less than conservative approach in assessing the data, discounting the possibility that a low power fuel rod may have failed. A "group-think" attitude prevailed that a fuel rod failure must demonstrate classical symptoms for it to exist. The team concluded that the licensee's fuel performance monitoring procedure was inadequate to detect a low power, peripheral fuel rod failure.

The licensee's "tramp uranium" explanation for the steadily increasing levels of fission products in the reactor coolant was reviewed by the team. It was noted by the licensee that the changes in coolant activity correlated with the increased delithiation process used for pH control. The team concluded that although the "tramp uranium" hypothesis was possible, it was improbable. For the licensee's hypothesis to be correct, fission isotopic increases should have been seen at the beginning of cycle 10 instead of later in the cycle. It would have been very unlikely for all of the "tramp uranium" to plate out at the beginning of the cycle and not be released until much later in the cycle.

The licensee's technical staff also failed to utilize their corporate NPAD health physicist, who was a known expert within the company, to evaluate the radiochemistry data. Late in the current outage, when this individual was on site for other reasons, he reviewed a hot particle personnel contamination event. He determined that this particle was composed of fission products and stated that this indicated a fuel failure, probably of a low power rod. He also noted in an informal memo to the Radiological Services Manager that he believed the observed fission product activity in smears, hot particles and air samples to be the result of clad failure of a least one fuel rod. This June 25, 1993, memo also stated his belief that the increase in fission product activities was from failed fuel and not from "tramp uranium." These conclusions were also documented on a performance assessment form dated June 26, 1993. The licensee also failed to act on this information, as refueling operations had nearly been completed, and the individual's assessment was based on a limited number of data points. However, this example further showed how the licensee tended to act in a less than conservative manner when presented with conflicting information and how available resources were not effectively used to resolve a confusing issue.

Given the evidence accumulated by the licensee's technical departments and support staff, the decision not to perform a more extensive fuel inspection of the peripheral assemblies at the end of cycle 10 was non-conservative. In addition, the licensee's fuel monitoring team did not make effective use of available in-house or external resources, did not pursue their assessment of fuel status, and failed to make a strong recommendation for fuel inspection to management. Plant management was aware of data which indicated a potential fuel failure, but was not aggressive in pressing the technical staff to adequately validate their conclusion that a fuel rod failure had not occurred.

3.6 I-24 Fuel Assembly Recovery

The licensee developed a procedure for removing the damaged I-24 assembly from the core. The licensee's Plant Review Committee (PRC) met on July 6 and reviewed and approved the procedure for use. The team questioned the adequacy of the procedure which was considered to be more of a guideline than a detailed procedure. Deficiencies with the procedure included insufficient details and precautions, inadequate guidance on what to do if problems arose, and insufficient senior reactor operator (SRO) signoffs. Independent of the NRC's review, the licensee's Nuclear Performance Assessment Department (NPAD) conducted a review of all fuel handling procedures as part of the prompt assessment activities required by the CAL. Based on feedback from the team and NPAD, the licensee revised the procedure and it was again reviewed and approved by the PRC on July 12. The revised procedure was considered satisfactory by the team.

The licensee's nuclear fuels group and reactor engineering staff made a presentation to the PRC concerning their plans to remove the I-24 assembly from the core. During this presentation, various options and alternatives were proposed. The PRC discussed and evaluated each option and a final approach was selected. The team considered the licensee's review to be thorough and their approach to planning the evolution satisfactory.

The pre-job briefing for this evolution was thorough and suitably detailed. The reactor engineering staff presented the overall purpose of the activity and the step by step details of the procedure. The presenters clearly stipulated that anyone could stop the activity if concerns arose, and that only the SRO on the bridge would be authorized to resume activities. The SRO was to be in total control. However, the team considered the location of the pre-job briefing, the licensee's Technical Support Center, to be inadequate due to its small size and insufficient seating capacity.

During the execution of the fuel assembly removal procedure, the reactor engineer and the SRO frequently discussed the execution and results of key steps. The approved procedure called for a thorough inspection of the top of the reactor with a video camera prior to moving any item. The examination of the I-24 assembly from above identified no abnormalities and the presence of the upper segment of the broken rod was verified. Various precautions, such as the camera scan of the I-24 assembly after removal of the adjacent M-36 assembly, were appropriate and minimized the risk of further damage.

Once the confidence in integrity of the assembly was established, the fuel bundle to the east of I-24 assembly, M-36 (see Figure 1), was removed. Removal of the M-36 assembly was marked by the occurrence of a hoist overload. This was verified to be the result of the top of the assembly catching on the edge of the hoist box, and the procedure to remedy this situation was properly executed. A camera was then used to monitor the full length of I-24 which allowed the portion of the assembly not hidden by the adjacent control rod blade (A-09) to be surveyed.

Based on the satisfactory appearance of the assembly, control rod blade A-09 was removed. Due to the potential for some binding because of torn spacer grids, there were precautions to monitor the override light on the console. Neither the override light nor any other abnormal indications were observed during the lift of control rod blade A-09. The camera was lowered again and a detailed examination of the I-24 assembly was performed. Contingencies were in place so that if there were any abnormal indications, all activities concerning the lift would cease. The initial examination of the I-24 assembly clearly revealed the presence of about 6-7 inches of the top of the corner rod and that it had shifted downward several inches. Review of the grid spacers showed significant damage in the corner region of all 10 grid spacers, including the total absence of material in some of the corners. The rods adjacent to the position where the failed S15 rod had been (R15 and S14), appeared to be firmly held by some portion of the spacers and the probability of dropping any other rods appeared remote.

The possibility of losing the remaining portion of the failed rod during the transfer operation was discussed. Since that portion of the rod contained no fuel and was only part of the cladding, the only concern was whether it would drop into the reactor. This rod segment apparently fell out and settled onto the core sometime during the lift and transfer evolution. The licensee's procedure stipulated that movement of the I-24 assembly over the reactor would be minimized; therefore, I-24 was moved in the manual vice semi-automatic mode of operation. Furthermore, both the lift and transport speeds, which are typically controlled semi-automatically, were accomplished in slow speed. The operator verified that after the lift of the I-24 assembly, he bypassed the reactor core totally. The movement of the I-24 appeared to take place smoothly. The rod segment was noticed to be missing as the assembly was being lowered into the tilt machine. The licensee terminated all activities to allow a fresh crew to continue recovery activities. The next crew located the dropped segment on top of assembly N-62 and retrieved it.

3.7 Post-Event Plans and Procedures

To justify the use of other than I-series assemblies, the licensee proposed to examine representative assemblies from groups of fuel assemblies that had been burned in the core for three cycles. This resulted in a number of detailed inspections involving first J and K-series assemblies, and then L-series assemblies. A number of fuel rods would be pulled from the selected assemblies to determine if excessive wear could be detected. If there were no significant indications, the licensee could reasonably conclude that these assemblies were suitable for a fourth cycle of irradiation. However, in order to ensure cladding integrity would exist during a fourth cycle of irradiation, the licensee examined a four-cycle burned H-series assembly. The H-series assemblies were identical in mechanical and nuclear design to I, J, and K-series assemblies, and nearly identical to the L-series assemblies, except for how the grid straps were manufactured. If the fuel rods of the H-series assembly showed no significant wear, the licensee intended to select a suitable combination of the assemblies under consideration. The licensee established an appropriate set of criteria for selecting usable fuel assemblies that took into account the salient characteristics of the fuel rod failure. Based on the licensee's proposal, the team concluded that their approach was reasonable and adequate. However, further review by the NRC of the licensee's safety evaluation of their final cycle 11 core load will be performed.

Licensee representatives estimated that approximately 219 fuel pellets were lost from one section of the broken fuel rod based on pellet length and the length of the empty broken rod piece. The two remaining sections of the rod did not appear to be missing any pellets. The concentration of fission isotopes in the reactor cavity area, along with the dose rates of filters used in vacuuming and cleaning up of the reactor cavity and tilt pit areas account for a small percentage of the missing fuel. If located, the licensee plans to recover the fuel although it is unlikely that these pellets would be intact due their age and burnup. The licensee stated that, according to SPC, these pellets would become "mushy" when exposed to water. Fuel recovery efforts were not complete when the team concluded its inspection. During the root cause analysis efforts, the licensee did solicit the assistance of an individual from Trojan Nuclear Plant, which had a similar fuel rod failure, to assist them in evaluating the potential effects of fuel in the primary coolant system during operation.

3.8 Root Cause Analysis

Determining the mechanism for the fuel rod failure proved to be a complex evolution involving many potential contributing factors. Identifying the root cause(s) of such a failure depends on first determining the failure mechanism. As discussed in Section 3.4, the team considered the failure mechanism to be influenced by a combination of operational, design, material, and mechanical factors. Other aspects such as procedural inadequacies and industry experience were also examined. The team evaluated the licensee's efforts in trying to determine each element's influence on the root cause and concluded that they were thorough and well developed. The status of the licensee's efforts in each of the various areas at the end of the inspection follows.

Operational factors: There are several operational factors which could have contributed to the failure. Of these, the team considered radiation exposure and primary coolant flow to be potentially significant factors. Core power density, local thermal hydraulic effects, and operational transients do not appear to have noticeably contributed to the fuel failure. The team assessed the licensee's evaluation of the effect of radiation exposure on the fuel, grid spacers, and the spacer springs. The licensee may be able to relate radiation exposure to the fuel failure, however, the analysis was not complete at the end of the inspection.

Initially, the licensee did not consider reactor coolant flow to be a major contributor to the fuel failure and members of the licensee's thermal hydraulics organization were not actively engaged in analyzing the failure mechanism. However, as the investigation progressed, flow effects were more prevalent in the licensee's root cause analysis efforts. The licensee requested their thermal hydraulics group, in concert with a vendor, to calculate differential pressure in the B-19 region of the core as well as similar locations. This included the shroud and core baffle areas. The licensee was also considering evaluating the potential effects of cross flow, jetting flow, and leakage flow in the core shroud areas adjacent to B-19 and other core periphery locations. Flow induced vibrations may have caused the corner of the fuel assembly to rub against the core shroud, creating excessive wear on the grid spacers and fuel rod. The assessment was not complete at the end of the inspection.

Design effects: Design features of the fuel assembly and its location in the core appeared to have played a major role in the fuel failure. The I-24 assembly was designed to operate without a failure up to 50,000 megawatt-days per metric ton (MWD/MT). The failed rod and the assembly had not exceeded 39,600 MWD/MT. SPC had approved the installation and operation of this assembly in that area of the core through cycle 11. However, the fuel apparently failed during cycle 10. Since more than one assembly that had been operated at a core shroud corner location exhibited signs of spacer fretting wear, it is apparent that this core location is a factor in causing the fuel rod failure.

Another key factor may be the grid spacer design and associated spring retention forces for I-series assemblies. As the grid spacers are irradiated, the material properties cause a preferential growth pattern, which in the case of the I-series assemblies results in an expansion of the grid spacer strap material horizontally. This manifests itself in growth in the corner areas of the fuel assemblies. This then leads to a reduction in retention force at this location. The corner rods are held in by only one spring and the grid strap, making it the least bounded rod in an assembly. Any looseness in this area will make the rod more susceptible to vibration and subsequent grid spacer fretting. SPC has improved the design of the spacer grid assemblies and the internal springs for more recent fuel assemblies (M-series and later). Since the licensee was aware that SPC had improved the design of their grid spacers to compensate for design weaknesses, then the I-series assemblies should have been scrutinized more thoroughly before placing them in the core for another cycle. This indicated a less than conservative approach taken by the licensee.

Material: Fuel assembly construction materials may be another key element in this failure. Cladding, spacers, and springs have unique material compositions and their performance in the operating conditions at the core shroud regions needs to be evaluated. The ability by either the clad or the spacers to better withstand the effects of vibration could have reduced the potential for this failure. The licensee assessed the effect of swelling of the Zircaloy-4 spacer with age and irradiation, relaxation of the Inconel-718 spring, fretting of the fuel rod and spacer, and wear of the spacers by the shroud. At the end of the inspection, a final determination had not been made.

Mechanical: There is a strong indication that there was a mechanical failure that caused the fuel rod to break into four segments and to tear away part of the cladding. The licensee evaluated the potential contribution of a mechanical interference during the lift of the I-24 assembly causing the failure. Initial indications were that the lift of the assembly might have played a significant part in breaking the fuel rod into four segments. This was supported by the appearance of green stick failures described in Section 3.4.

One likely scenario is that a break in the fuel rod already existed, and when the I-24 assembly was lifted out of the tilt machine, the fuel rod did not rise up into the hoist box along with the rest of the assembly. Instead, because it was extended outside the assembly due to the absence of spacer grid strap material in the corner, it rose up in a region between the hoist box and the mast, eventually catching on a protruding object within the mast, snapping the fuel rod in several other places. Another possible scenario is that the rod was protruding from the assembly as it was moved in and out of the tilt machine, and that it caught on one of the structural members of the equipment. This resulted in the rod breaking into several segments. The team considered such events to be plausible.

Equipment: Other than the transport equipment used to transfer the fuel within the reactor cavity, there was also an ultrasonic test rig used at the end of cycle 9 to test all the fuel assemblies, including I-24. The licensee's investigation included the potential impact of this equipment. Although the effect of this equipment on the failure was discounted, this was representative of the licensee's comprehensive evaluation of pertinent factors.

Operator Error: No operator error related to this failure (during fuel handling, inspection, or testing) surfaced during this inspection. Although operator error did not appear likely, the licensee's evaluation included this as a potential contributing element. The team concluded that the damage to the fuel rod could have occurred during fuel handling activities without attracting operator attention.

Training and Qualifications: The fuel handling operations are conducted by licensed operators. Conduct of fuel handling activities was included in the operator continuing training program. The operators took a refresher course in fuel handling prior to this refueling outage. Any changes in Procedure FHSO-2, "Refueling Procedure," since the last refueling outage were reviewed. Observations of fuel handling activities by the team indicated that the operators understood their equipment and procedures, and were well-qualified to operate the refueling machines. The licensee's training program for fuel handling appeared adequate.

Procedures: The team questioned the adequacy of Procedure RSA-03, "Fuel Performance Monitoring," Revision 3, as noted in Section 3.5. The licensee's staff agreed that this procedure did not appear to be adequate in determining the presence of low power, peripheral fuel rod failures, and committed to improve the procedure and increase its sensitivity for detecting similar events. As part of their prompt assessment, the licensee's NPAD organization evaluated the adequacy of fuel handling procedures. NPAD noted that the procedures were vague in describing interlock usage and to what extent the refueling machine could be moved when interlocks were bypassed. They also noted that logging requirements needed to be upgraded such that incidents of hoist underloads, overloads, and bypass key usage were documented. The procedures were upgraded prior to recommencing refueling activities on July 12, 1993. The licensee also committed to doing a review of refueling procedures after the outage to incorporate additional feedback.

Industry Experience: From indications, spacer fretting wear appeared to occur. This phenomenon had been prevalent in PWRs in the past, particularly Westinghouse plants until they made a design modification. Fretting had been one of the primary causes of fuel failure. The licensee contended that the CE design did not have the same flow characteristics as Westinghouse around the shroud area. Therefore, they did not aggressively pursue evaluating existing data on these failures to assist them in conducting a root cause analysis. They relied on SPC's analysis and experience to assist them in conducting the bulk of the root cause analysis.

Metallurgical Examination: The team believed that a metallurgical examination of the failed fuel rod would be beneficial and could be used to confirm or reject the conclusions drawn in the root cause analysis. The metallurgical examination could establish the initiation site for each fracture, the nature of the fracture, and if hydriding was present. At the end of the inspection, the licensee had not yet determined whether such an examination would be performed.

3.9 Corrective Actions

3.9.1 Immediate Corrective Actions

When the high radiation levels in the reactor cavity tilt pit area were discovered, the licensee immediately halted activities to conduct a radiological assessment. The hazard to plant personnel in containment was minimized by raising the level of water in the tilt pit. Containment integrity was established prior to handling the broken fuel rod, which showed a sensitivity towards maintaining public health and safety. The licensee communicated with the NRC on a daily basis, keeping the agency well-informed of plant status. The team considered the licensee's immediate actions to be prudent and safe.

3.9.2 Pre-Startup Corrective Actions

Final corrective actions will be developed when the root cause investigation is complete. However, the licensee took some interim actions based on their current evaluation efforts. Due to significant fretting wear noted on several I-series assemblies, the licensee decided to remove all the I-series assemblies from the cycle 11 core and set out to determine suitable replacement assemblies. The licensee determined that it would not use fuel assemblies with four or more cycles of operation. This decision was based mainly on cycle 9 ultrasonic test results which found no defects on assemblies that had been operated for four cycles. The licensee established pertinent acceptance criteria for selecting the assemblies, which the team found to be appropriate.

Use of stainless steel rods were being considered for use in core shroud corner locations. The licensee committed to provide the NRC information regarding their proposed core loading pattern and the rationale for using the assemblies that replace the I-series assemblies.

Concerns over the unaccounted for debris and fuel pellets still existed at the end of the inspection. The licensee committed to perform a visual inspection underneath the core support plate to try and identify and recover any material that may have deposited there. Further inspections of the tilt pit were also planned as the licensee continued to identify and recover as much of the cladding and fuel material as possible. Though the final plans and procedures were not available for review by the team, the licensee appeared to be proceeding in a cautious and thorough manner.

3.9.3 Post-Startup Corrective Actions

As a long-term corrective action, the licensee had already ordered 16 high thermal performance P-series assemblies to replace the I-series assemblies for cycle 12. These fuel assemblies will contain 5 rows of AISI Type 304L stainless steel solid rods on the side of the assembly next to the core shroud as part of the flux reduction program. The remaining rods will be low enrichment fuel rods. The 8 guide rods located on the peripheral rows of the fuel assembly will also be 304L stainless steel. The spacers have been redesigned for these fuel assemblies. The top and bottom spacers will be 2 1/2 inch, high thermal performance, bi-metallic spacers. The 8 intermediate spacers will be 2 inch Zircaloy-4 spacers containing no Inconel springs. The fuel rods will be held in place using Zircaloy-4 springs that are integral with the spacer strips. These assemblies have been designed for 6 cycles of operation without being removed from the core during refueling outages.

Radiochemistry personnel will monitor plant systems including primary coolant and chemical volume control system (CVCS) filters for debris and fission isotopes following plant start up. Any loose fuel left in the reactor vessel could result in significant increases in fission isotopes in primary coolant, including noble gases and iodines. This would make it more difficult to detect fuel failures, particularly those occurring in low power rods. Also, hot particles could spread throughout the primary coolant system and result in an increased number of personnel contamination events due to fission products during future outages. Deposition of these fission products in plant systems would result in increased radiation levels for future maintenance activities and impact the ALARA program. At the end of the inspection, the licensee was developing plans for monitoring the plant for the presence of loose parts and a possible increase in fission product activity.

3.10 Management Involvement

During the inspection, the team noted a high level of management involvement that fostered an effective, team-oriented approach towards problem solving. With the increased management attention, deliberations by staff members were appropriately cautious and thorough. Management allowed the staff the latitude to make recommendations and provide options without overt direction. Management decisions were made with a reasoned approach and adequate safety considerations. There were no appearances of excessive management pressure to perform any refueling activities or to influence the results of root cause analysis efforts.

3.11 Safety Assessment

The NRC staff approved the licensee's cycle 9 reload in a safety evaluation dated February 20, 1991. The I-series assemblies were inserted in the cycle 9 core peripheral locations for fluence reduction purposes. For cycle 10 and cycle 11 safety analyses, the licensee concluded that there were no unreviewed safety questions in accordance with 10 CFR 50.59. The safety analyses covered mechanical design, nuclear design, and thermal hydraulic analysis. The cycle 10 and initial cycle 11 cores were very similar to the cycle 9 core. The I-series assemblies were in their fifth cycle of operation during cycle 10, and would have been in their sixth cycle in cycle 11. The I-series assembly locations were not changed from cycle 10 to cycle 11 except for being rotated 180 degrees to compensate for the bowing of fuel assemblies with increased exposure. The licensee justified the continued operation and high exposure of I-series assemblies based on their analysis of similar J and K fuel assemblies. The team reviewed the licensee's cycle 10 and cycle 11 safety analyses and given the information that was available, concurred with the licensee's assessment that the I-series assemblies were suitable for use.

However, the licensee's safety evaluations did not include an analysis of the effect of fast fluence on the mechanical components of the I-Series assemblies, particularly the grid spacers and spacer springs. This resulted in the potential effects on spacer spring retention forces being unanalyzed. Placing the hafnium rods in the I-series assemblies significantly reduced the thermal flux to which they were exposed. However, it did not significantly change the fast flux. Therefore, these assemblies were used in conditions that were different from their intended design. The team concluded that the licensee's 10 CFR 50.59 analyses were therefore inadequate with respect to evaluating the mechanical properties of the fuel assemblies. The revised cycle 11 core configuration will require an additional safety evaluation by the licensee. The NRC intends to review this safety evaluation to determine its adequacy.

3.12 Generic Implications

The I-24 assembly was located on the periphery of the core for fluence reduction purposes. The five actual (and intended six) cycles of operation was atypical for SPC PWR fuel. SPC has more extensive experience with fuel assemblies that have been through five and more cycles of operation in boiling water reactors (BWRs). Use of SPC fuel of this design for five or more cycles may cause spacer growth which results in spring relaxation in the corner regions of the assembly. This sets up a situation where spacer fretting and subsequent fuel failures can occur. SPC recently modified the manufacturing process for spacer grids so that the preferential growth pattern minimized this phenomenon. However, there remains the concern for the acceptability of using fuel assemblies of similar design and vintage to the I-series assemblies for five or more cycles of operation.

During cycle 10 operation, low level fission product activity in the coolant gradually increased such that it may have indicated a fuel failure. However, traditional symptoms, e.g., increased noble gases and iodine spiking, did not appear. The team has generic concerns that commonly applied fuel failure detection methods are inappropriate for older, low power, peripheral assemblies in pressurized water reactors (PWRs).

3.13 Core Design Responsibility and Outside Support

The initial core design of the facility was developed by Combustion Engineering (CE). In addition to the initial core design, CF supplied the first core load. SPC (previously Advanced Nuclear Fuels) supplied the fuel from the first reload through the present. All the fuel used in the core since cycle four has been supplied by SPC. Bechtel Power Company, the architect/engineering firm which designed the plant, and Westinghouse Electric Company have all been used to perform some of the design basis calculations. There have also been a number of other vendors who have been used to a limited extent for specific analyses and other support.

In general, the licensee staff recognized and maintained overall responsibility for the core and the fuel. However, they relied heavily on SPC to provide support. This support ranged from the reload core design to design basis calculations and analyses associated with thermal hydraulic evaluations and events contained in Chapter 15 of the FSAR. Within the last several years CE's role has gradually been diminished while SPC's role increased. However, analysis of some of the Chapter 15 events were still performed by CE. The licensee appeared satisfied with the support provided by SPC. They audit SPC's manufacturing facilities at least once a year and provide comments to SPC in a report. The team reviewed several of the most recent reports and noted that the licensee appears to be conducting audits that are of appropriate scope and depth. However, the team also made the observation that the licensee needed to be more aggressive in ensuring open issues are resolved.

4.0 Stuck Fuel Assembly Event

4.1 Summary of Events

On September 3, 1988, while the upper guide structure (UGS) was being removed from the reactor vessel, a fuel assembly (K-28) at core location Z-11 was observed to be hanging from the bottom of the UGS. At the time the stuck assembly was noticed, the bottom of the assembly was 15 inches above the reactor vessel flange. An Unusual Event was declared. The stuck assembly K-28 was grappled with the use of "j-hooks" through the UGS flow holes. It was then separated from the UGS by applying a vertical force with a slide hammer to the assembly's upper tie plate and carefully lowered onto the top of the core. The UGS was placed in its normal reactor cavity storage position and K-28 was transferred from the reactor vessel to the spent fuel pool. The Unusual Event was terminated on September 7, 1988.

On February 29, 1992, a similar event occurred when removing the UGS from the reactor vessel. A fuel assembly (I-28) at location Z-11 was observed to be hanging from the bottom of the UGS. The stuck assembly was not noticed until the bottom of the assembly was lifted 2 feet above the reactor vessel flange. An Unusual Event was declared. The stuck assembly was grappled with the use of a "j-hook" through the UGS flow holes. A second "j-hook" was attached to I-28, the tension on the first "j-hook" was removed, and this caused the fuel assembly to separate from the UGS. The assembly was then carefully lowered onto the top of the core. The UGS was placed in its normal reactor cavity storage position and I-28 was transferred to the spent fuel pool. The unusual event was terminated on March 3, 1992.

On July 6, 1993, after the UGS had been lifted of three feet, a fuel assembly (SAN-8) was observed, by camera, to be attached to the bottom of the UGS at the Z-11 location. An Unusual Event was declared. On July 7, unsuccessful attempts were made to disengage the SAN-8 assembly from the UGS by alternately adjusting the tension on the two chainfalls which were attached to the assembly by "j-hooks". The licensee also made an unsuccessful attempt to free the assembly by performing Step 5.2.9 in Procedure No. FHSO-18, Rev. 0, "Recovery of Assembly SAN-8." This was also unsuccessful because the slide hammer was too heavy to effectively manipulate. On July 8, a modified slide hammer was used to apply a striking force several times to the SAN-8 upper tie plate. Assembly SAN-8 was freed from the UGS and carefully lowered into core location Z-11. The UGS was placed on its storage pads. The Unusual Event was terminated. The team reviewed the completed procedure and interviewed the personnel involved with the evolution. The team concluded that although the lift operation did not appear to cause the stuck assembly, a number of work performance problems were identified (See Section 4.4).

4.2 Description of Equipment

Upper Guide Structure: The upper guide structure (UGS) consists of a flanged grid structure, 45 control rod shrouds, a fuel assembly alignment plate, and a ring rim (See Figure 8). The UGS aligns and supports the upper end (tie plate) of the fuel assemblies, maintains the control rod channel spacing, prevents the assemblies from being lifted out of position during a severe accident condition, and protects the control rods from the effect of coolant cross flow in the upper plenum. The UGS weighs approximately 57,000 pounds and is approximately 140 inches in both height and diameter. The fuel assembly alignment plate is designed to align the upper tie plates of the fuel assemblies. Each assembly is aligned by inserting the two diagonally positioned UGS fuel assembly alignment pins (See Figure 9) into the corresponding holes in the assembly upper tie plate (See Figure 10). All three stuck assembly events occurred at core location Z-11.

Fuel Assembly: See Section 3.2 of this report.

UGS Lifting Equipment: Both the UGS removal and installation were performed by a contractor, Westinghouse Refueling Services (WRS), using the polar crane with attached load links (See Figure 11) and the lift rig (See Figure 12). The lift rig was attached to the UGS upper flange at three equally spaced attachment points. The UGS was under water during removal and storage. Upon removal of the UGS from the core support barrel, the UGS was raised to a height to clear any possible obstacles and moved to the UGS laydown area and placed on the three storage pads (See Figure 13).

4.3 Detailed Sequence of Events (7/6/93 - 7/8/93)

Attachment G provides a detailed sequence of events for the lifting of the stuck fuel assembly that occurred on July 6 until it was safely released and lowered into the core on July 8.

4.4 Evaluation of Events

4.4.1 July 6 Stuck Assembly Event

Fuel assembly SAN-8 was stuck to the upper guide structure (UGS) and raised 3 feet from its resting core location, Z-11, while the UGS was being lifted. The problems with personnel performance, procedural adherence, attention to detail, and work controls and practices described below indicated that management involvement was inadequate to provide proper control over the UGS lift.

Personnel Performance

During the pre-job briefing, the presentation by the refueling contractor (WRS) did not indicate the intent to use new equipment (J-300 load cell readout device) for load monitoring. The Senior Reactor Operator (SRO) was not made aware of this intention until he entered the containment for the UGS lift. Although this new equipment was not yet incorporated into the procedure, the SRO failed to correct this situation by stopping the work. This is an example of an inadequate pre-job briefing and inadequate contractor control.

Procedure Quality and Adherence

Procedure No. RVI-M-1, Revision 16, dated 6/18/93, "Removal and Storage of the Upper Guide Structure," as revised by Temporary Change No. M93-038 (Work Order No. 24302226, 7/6/93), was being performed on 7/6/93 when assembly SAN-8 was discovered to be stuck to the UGS. Although not directly responsible for the stuck assembly, the following problems with procedure quality and adherence occurred during the UGS lift:

- The refueling contractor added new equipment (Westinghouse load cell readout device, J-300), which had not yet been incorporated into the procedure, to the calibrated equipment listed in Section 3.6.1.

- The prescribed steps in Sections 5.3.6.a through 5.3.6.f were not followed for setup of the approved load cell readout device, TI-2000. The procedure change process was not used to include the operational instructions for the Westinghouse J-300 load cell readout device, which was actually used.

- Although Section 5.3.6.g specifies Work Order No. 24301781 for the steps required to use both the approved TI-2000 load cell readout device and the J-300 readout device actually used, WRS failed to follow work order Step 3.3.A.7 to zero the readout device. This resulted in erroneous UGS weight readouts which were 6,800 pounds too high.

- While lifting the UGS, WRS failed to adhere to the requirements of Section 5.3.14, therefore, exceeding the prescribed load cell upper limit of 62,000 pounds.

Attention to Detail

The failure to zero the J-300 load readout device led to erroneously high UGS load indications. This resulted in the inability of the WRS Supervisor to obtain correct load readings. Although only one fuel assembly was lifted and there were no visible indications that the UGS lift rig was experiencing significant binding between the UGS fuel alignment pins and the fuel assembly upper tie plate holes, the J-300 load cell readout indicated binding in excess of the maximum expected by the lifting procedure. This false indication was due to a lack of attention to detail in that the load cell readout device was not zeroed prior to use.

Work Controls and Practices

Procedure RVI-M-1 established a maximum apparent weight of 62,000 pounds for the UGS lift. This value was derived from a UGS weight of 55,800 pounds plus approximately a 10% load for potential breakaway forces (binding force between the UGS and the fuel assemblies). The WRS Supervisor observed the load cell readout approaching 60,000 pounds with no movement of the UGS. Not realizing that the load cell was initially reading 6,800 pounds too high, the supervisor concluded that there could be significant binding and signaled the crane director to stop the lift. The crane director signaled the crane signalman to stop the lift. The signalman signaled the crane operator to stop the lift. By the time the crane operator stopped the lift, the UGS (and stuck assembly SAN-8) had been raised about 6 inches and the load cell read 62,800 pounds.

This sequence of communication for control of the lift was normal for conducting the UGS lift. The awkward communications coupled with the slow response of the crane made lift cessation untimely and resulted in exceeding the procedural maximum expected reading. Camera observations showed that only one fuel assembly was attached to the UGS. The load cell reading (when corrected for the failure to zero the readout device) showed no excessive load beyond the increase due to the weight of the assembly. Poor work controls and practices led to the inability of the work crew to have positive and effective control of the UGS lift as prescribed by the procedure.

The polar crane used to align the UGS with both the reactor vessel and the storage pads was positioned based on markings on the crane arms and the crane rail, which are approximately 50 feet above the refueling floor. To position the UGS onto the reactor vessel or the storage pads, the refueling contractor visually checked these markings, and visually sighted the reactor vessel guide pins or the UGS storage pads, which are under approximately 20 feet of water. These difficult visual checks have significant potential for human error.

Management Involvement

Because the licensee experienced some problems with refueling contractor performance during the 1992 refueling outage, the performance expectations were more clearly emphasized to contractor personnel for the 1993 refueling outage. These were delineated to the contractor in a special training and qualification program, written performance guidelines, and written procedural performance expectations for refueling services contractors. However, as noted above, the licensee did not adequately control WRS.

4.4.2 July 7 Unsuccessful Attempt to Free Stuck Assembly

Unsuccessful attempts were made to disengage the SAN-8 assembly from the UGS by alternately adjusting the tension on the two chainfalls. Procedure No. FHSO-18, Rev. 0, "Recovery of Bundle SAN-8," step 5.2.9, could not be performed because the slide hammer was too heavy to handle. Management control over refueling activities remained inadequate as there were problems with personnel performance, attention to detail, work control and practices, and procedure adherence.

Personnel Performance/Attention to Detail

During the pre-job briefing for the July 7 SAN-8 assembly recovery effort, the WRS Supervisor appeared to be overconfident and did not stress the importance of heightened safety awareness and lessons learned from previous similar events.

Examples of deficient industrial safety awareness were observed during the attempt to disengage the SAN-8 assembly from the UGS. The refueling contractor supervisor stepped on the lower railing of the catwalk to gain additional distance in reaching the UGS lifting rig without a safety harness. Two refueling contractor personnel attempting to remove the heavy slide hammer from the reactor cavity were positioned between the safety railing and the reactor cavity without safety harnesses. Also observed was the removal of procedurally required safety glasses by the contractor while still conducting work.

Work Controls and Practices

The noise from the containment ventilation fans prevented normal voice communication unless it was conducted face-to-face. There was virtually no effort by the refueling contractors to communicate with the SRO supervisor while various attempts were made to attach the hooks and the chainfalls to the SAN-8 assembly. The SRO did correct the contractor once when the contractor mistakenly thought that a hook was attached to the SAN-8 assembly.

Procedure Adherence

Steps 4.2.6 and 5.2.1 in Procedure No. FHSO-18, "Recovery of Bundle SAN-8," Rev. 0, provided direction to tighten the chainfalls to a combined maximum load of 1500-1600 pounds. The refueling contractor executing the procedure reached a combined load of 2300 pounds.

Management Involvement

During the attempts to disengage the stuck fuel assembly, the NRC inspectors observed the refueling contract workers leaning over railings, removing safety glasses, and conducting work without informing the licensee's supervisor of their intentions. It was clear that the licensee supervisor was not controlling the work performed by the contractors. The supervisor did not intercede until the refueling contract workers had exceeded the procedural limits on the chainfall loading.

4.4.3 July 8 Release of Stuck Assembly

After applying a striking force from the new slide hammer several times to the upper tie plate, assembly SAN-8 was released from the UGS alignment pins and was lowered back into core location Z-11. The UGS was placed on its storage pads. Although there were still some problems with procedural adequacy, personnel performance, and attention to detail, increased management involvement improved refueling contractor performance.

Personnel Performance/Attention to Detail

The pre-job briefing conducted by the licensee and their refueling contractor covered detailed steps of the evolution and the safety measures needed for the job. The refueling contractor personnel properly followed procedures to recover the SAN-8 assembly, lift the UGS to the required elevation, traverse it over the UGS storage pads and successfully set it down. During the movement of the UGS, the UGS upper flange bumped the ladder cage at the northeast corner of the reactor cavity (this was not in the direct path between the reactor vessel and the UGS laydown area). A slight dent was observed on the ladder cage and there was no observed damage to the UGS. This was documented in the licensee's Deviation Report D-PAL-93-145 and a root cause analysis will be performed by the licensee.

Procedural Adequacy

Revision 1 of Procedure No. FHSO-18, "Recovery of Bundle SAN-8," which included the steps needed to use the improved slide hammer, was used to successfully disengage SAN-8 from the UGS alignment pins. Revision 16 of Procedure No. RVI-M-1, "Removal and Storage of the UGS," was used to move the UGS to its storage pads. Step 5.3.17 directed the crane operator to raise the UGS until the bottom of the center horizontal rail on the lower platform of the UGS lift rig was level with the floor of the 649' elevation. Attachment 3 of the procedure provided a picture of the lift rig for use by the operators.

In the procedure, the top horizontal handrail depicted in Figure 12 was incorrectly identified as the "center horizontal handrail". This mistake could lead the crane operator to raise the lift rig to a lower elevation than necessary to clear the obstacles prior to traversing the UGS.

Work Controls and Management Involvement

The licensee SRO was in close proximity to the refueling contractor supervisor throughout the entire evolution which facilitated communications between the licensee and the contractors. This enabled the licensee management to provide close supervision of the contractors and thus positively control the contractor's performance.

4.5 Root Cause Analysis

The licensee organized a UGS Root Cause Team (UGSRCT) for the stuck assembly event to investigate the circumstances surrounding this event and determine its causes. The UGSRCT, with multi-discipline backgrounds, consisted of both licensee personnel and outside contractors.

This was the third stuck fuel assembly event at the same core location and two previous root cause analyses had been performed in 1988 and 1992. For both of these analyses the root cause was inconclusive. The UGSRCT developed a root cause analysis plan for the 1993 event which built on the licensee's previous analyses and which identified several additional potential contributors to the event. The following potential causes were under consideration for the three stuck assembly events. Each potential cause in annotated to indicate the year(s) in which it was considered in the root cause analysis.

- The UGS alignment pins at the Z-11 core location were bent causing the pins to be forced into the fuel assembly upper tie plate holes. (1988, 1992, 1993)

- Debris on the UGS alignment pins or fuel assembly upper tie plate holes caused a misalignment. (1988, 1992, 1993)

- Manufacturing deviation on the fuel assembly upper tie plate holes. (1988, 1992, 1993)

- A tolerance stack-up issue that could cause the UGS alignment pins to be forced into the fuel assembly upper tie plate holes. (1988, 1992, 1993)

- Debris on the core support plate at the Z-11 location could cause misalignment of the fuel assembly. (1988, 1992, 1993)

- Problems with the UGS lift technique or lift rig levelness. (1992, 1993)

- Deformation of the core shroud. (1993)

- Fuel assembly bowing. (1993)

- Core barrel mislocated. (1993)

- Damage to the UGS alignment pins or to the UGS lower alignment plate. (1993)

- Degraded surface condition of the UGS alignment pins. (1993)

- Loss of preload on the cap screws which hold the UGS together and could cause a significant loss of structural integrity. (1993)

The team reviewed the root cause analyses performed for the first two events and the ongoing analysis for the 1993 event. The previous analyses reviewed a wide variety of issues and appeared to sufficiently evaluate each of the possible causes, although an ultimate root cause was not identified for either event. The root cause analysis plan for the 1993 event appeared more formal and was reviewed and approved by licensee upper management. Action items were assigned to UGSRCT members for analysis and discussed at meetings to review the progress of each of the potential contributors. The team concluded that the investigations and analyses for the previous events and in progress for the present event were adequate.

4.6 Implementation of Previous Corrective Actions

The licensee instituted a number of corrective actions in an attempt to prevent the recurrence of the stuck assembly after both the 1988 and 1992 events. Some of these actions specifically addressed recurrence of the stuck fuel assembly, while others addressed problems identified during the performance of UGS lifts. Several of the corrective actions are discussed below, including their effectiveness in addressing the specific problem.

After the 1988 stuck assembly event, the licensee determined that a camera inspection should be performed when the UGS was 3 feet above the reactor core to determine if a fuel assembly was stuck to the bottom of the UGS, while the assembly was still being supported by the other fuel assemblies in the core. Although a camera inspection was performed during the 1992 UGS lift, the inspection was ineffective since the stuck assembly went unnoticed. The camera inspection looked at the top surface of the core and not the underside of the UGS, which made identifying the stuck assembly difficult. In addition, the UGS lift procedure identified the camera inspection as optional and not a requirement. Positioning spotters to look for stuck assemblies was also ineffective. The stuck assembly was not identified until the SRO on the refueling floor noticed that it was completely out of the core. After reviewing this event, the licensee considered that the procedure and pre-job brief guidance to the spotters might have been insufficient in that the location of the 1988 stuck assembly was not specified. As a result, the action taken to promptly identify a stuck assembly were ineffective during the 1992 refueling outage.

The UGS lift procedure was subsequently revised to include a requirement for the camera inspection to focus on the underside of the UGS and provide additional guidance on where possible stuck fuel assemblies might occur.

During the second 1993 UGS lift, the camera inspection clearly identified the stuck assembly when the UGS was 3 feet above the reactor core. The corrective action was effective in identifying a stuck assembly prior to it being completely removed from the reactor core.

One of the corrective actions identified after the 1988 event was that a more precise load cell readout device should be used during the lift of the UGS which might aid in recognizing a stuck assembly. A Model TI-2000 load cell readout device was procured and used during the 1990 UGS reinstallation to determine the weight of the UGS to be incorporated into the UGS lift procedure. Due to the tolerances in the accuracy of the readout device, the UGS lift procedure stated that the weight of a stuck fuel assembly may not be evident even with this equipment. The procedure used for the 1992 UGS lift had not been revised to include the weight of the UGS or to state the specific load cell readout device to be used. The licensee used the 300,000 lb polar crane load cell readout device which was not calibrated, nor precise enough for the UGS lift. The team concluded that the 1988 corrective actions concerning the load cell were not adequately implemented in 1992.

The UGS lift procedure used in 1993 specified the use of the TI-2000 load cell readout device and the expected weight of the UGS. A maximum weight was also included in the procedure. The readout device actually used, however, was a Westinghouse model J-300 that was more precise than the TI-2000, but was not specified in the procedure. No procedure change was made to allow the use of this new readout device. In addition, the load cell readout device was not zeroed to take into account the weight of the load link assembly. By not zeroing the readout device, it appeared that the maximum force allowed by the procedure (62,000 lbs.) for the UGS lift was exceeded by 800 lbs. The failure to comply with the procedure by the refueling crew was not identified by the SRO in containment, who had overall control of core alterations, nor a Nuclear Performance Assessment Department (NPAD) assessor who was field monitoring this work. Procedural adherence was a problem identified during the June 15, 1993, control blade lift event and was to have been corrected for the 1993 UGS lift. The corrective actions implemented were not effective in preventing procedural non-compliance.

As a result of the previous stuck assembly events, the UGS lift procedure was revised to include management hold points at important steps to ensure that any actions taken were appropriate and that no unusual conditions existed prior to continuing with the procedure. These changes were effective as the stuck assembly was detected early in the lift procedure during the 1993 event and it helped to prevent pulling a fuel assembly completely out of the core.

The UGS lift procedure in 1988 required that the UGS be raised 12 inches above the reactor cavity floor prior to moving the UGS. This was to prevent damage due to bumping obstacles on the cavity floor. The UGS storage pads, however, stand 14.5 inches above the cavity floor, which creates an obstacle where damage to the UGS pins could occur. After the 1988 event, the licensee performed a study which determined that the UGS needed to be raised such that the bottom of the center horizontal rail on the lower lift rig platform was level with the refueling floor (649' level) to ensure that it would clear all obstacles in the reactor cavity. This would provide a 13 inch clearance over the UGS storage pads. Based on the procedure in effect in 1988 and before, the possibility existed that damage to the UGS alignment pins could occur while moving the UGS to the UGS storage pad. Even though the procedure was changed after 1988, there are some other aspects of moving the UGS that could result in inadvertently bumping the alignment pins. For example, the verification that the UGS cleared all obstacles in the reactor cavity is done visually and may not be accurate. In addition, if the polar crane is not aligned precisely over the UGS when preparing to lift the UGS off of its storage pads, the possibility exists for the UGS to swing horizontally and bump the storage pad located in close proximity to the Z-11 alignment pins.

In 1988, the UGS alignment pins at core location Z-11 were determined to be essentially straight when gauged by CE. However, based on records that stated the gauge did not completely fit over the pins, it can be concluded that the pins were most likely bent by 1988. Calculations from 1992 concluded that minor pin bending (<0.7°) could cause the UGS alignment pins to be forced into the UTP holes. The Z-11 pins were gauged in 1992 by Westinghouse and determined to have bend angles of 1.145° and 0.548°. These pins were straightened prior to plant restart. After the stuck assembly in 1993, the pins were gauged and determined to have bend angles of 1.56° and 0.41°. Two impact or peen marks were found on the SAN-8 assembly's UTP. These were apparently caused by the UGS alignment pins striking the SAN-8 UTP and most likely occurred during the reinstallation of the UGS from the current refueling outage. The location of these marks, next to the UTP holes, showed that there was a misalignment between the pins and the holes of approximately 0.5 inches. At the end of the inspection, the licensee was evaluating the cause for this misalignment and determining how the pins were bent. Since the inspection, the pins have been replaced.

A tolerance stack up study performed in 1988 indicated that a potential interference fit (1 mil) could exist based on a worst case scenario for the UGS alignment pins and the UTP holes. As a result, the UTP hole diameter for fuel assemblies starting with the M-series was increased 2 mils. The tolerance stack up study performed in 1992 noted that an error existed in the 1988 study and no interference fit should exist. The licensee replaced the SAN-8 UTP and was considering increasing the UTP hole tolerance for future reloads fuel assemblies.

The team concluded that the licensee did not thoroughly implement several corrective actions taken as a result of the previous stuck assembly events and other fuel handling-related refueling problems and were therefore ineffective in correcting the identified concerns. Of particular note was the problem with adhering to procedures which was also noted in the June 15, 1993, control rod uncoupling event.

4.7 Safety Assessment

The stuck fuel assembly event had no consequences for public health and safety. Since it was discovered after being lifted only three feet from the core, the risk to personnel was minimized and very small. The licensee also took steps to minimize the risk to personnel by closing off the containment and limiting access while the assembly was stuck. There was risk to personnel performing work when good industrial safety practices were not followed.

The safety significance of this event is the licensee's inability to effectively implement corrective actions. This event demonstrated instances where corrective actions were not comprehensive enough, corrective actions were not followed up or thoroughly implemented and personnel performance problems continued.

5.0 Licensee Self Assessment

5.1 Prompt Assessment

As required by paragraph 3 of the Confirmatory Action Letter, the licensee conducted a prompt assessment of the refueling events that occurred during the outage and identified any near term corrective actions needed to resume refueling activities. The assessment was conducted by the licensee's Nuclear Performance Assessment Department (NPAD) which reviewed recent operator errors, refueling events and previous NPAD assessments. The results were reviewed with the team. NPAD concluded that error rates from 1992 to 1993 had not increased and no generic training faults existed. NPAD also concluded that a review of refueling procedures was necessary and that corrective actions for recent events were too specific and needed to be broadened. Actions which resulted from the prompt assessment were: the assignment of senior managers on shift to oversee refueling activities, implementation of a pre-job brief checklist, a management plan for the refueling restart, review and revision of refueling procedures prior to use, and specific clarifications and refinements of fuel handling practices. After discussing the prompt assessment findings and the implementation of the corrective actions, the team agreed on July 12 that refueling activities could proceed. The team considered NPAD's prompt assessment to be thorough and self-critical, resulting in several significant corrective actions.

5.2 Quality Assurance Activities

Even though the NPAD was tracking corrective actions associated with the previous refueling outages, it was not effective in identifying the problems that were noted during the control rod uncoupling event or many of the issues identified by the team during its inspection.

NPAD was actively involved in the resolution of open issues identified in the last outage, including contractor adherence to administrative controls. It assessed preparations for the 1993 outage including a review of the status of previously identified corrective actions. Although determining the effectiveness of these corrective actions was not part of the assessment, the report contained several overly optimistic conclusions. This suggested to the team that the NPAD staff may be too involved in resolving issues rather than independently evaluating the effectiveness of corrective actions and identifying potential problems.

In early 1993, the NPAD organization began a program of field monitoring in which assessors observe work activities and generate performance assessment forms (brief field monitoring reports) which grade performance. The results of all the reports were summarized weekly and quarterly. NPAD was aware of previous outage problems with the Westinghouse Refueling Services (WRS) contractor and led an effort to clarify expectations between the contractor and the operations department. This included providing training to each of the operating shifts. A summary of the NPAD field monitoring of WRS activities showed 45 hours of observation by 7 different assessors. No significant problems with procedure adherence were identified even though 6 hours of observation were of the UGS lift on July 6, 1993. NPAD generated a deviation report on July 14, 1993, for the procedural adherence problems involved with the lift after it was identified by the team.

A notable NPAD monitoring report, as discussed in Section 3.5, was generated by an experienced health physics assessor which postulated the presence of a low power assembly fuel failure. This report was dated June 26, 1993, which was concurrent with reactor vessel head reinstallation activities. This report was not considered conclusive by plant management and was not acted upon. NPAD missed an opportunity to direct management to a more conservative course of action.

6.0 Conclusions

The team made three broad conclusions as a result of the inspection. These are summarized below:

1) Management expectations for performance were not being effectively translated to the working levels on a consistent basis. The licensee instituted a philosophy of empowerment that had not yet been fully implemented. It was the team's impression that there was a lack of commitment at the working level and this prevented its effective facilitation. Evidence of this was seen in the repetitious nature of many of the team's observations.

During the inspection, the team noted a concerted effort by licensee management to ensure its expectations were conveyed to the staff through the working level. This effort was encouraging, however, this demonstrated that perhaps a greater level of management attention will be required until the commitment to quality is ingrained throughout the organization. The team also noted that additional training for first line supervisors to assist them in implementing managerial goals, may be warranted.

2) The team had concerns regarding the licensee's ability to make conservative decisions regarding the operation of Palisades. There were several examples where the licensee was presented conflicting evidence and opinions that indicated a more conservative approach might have been warranted. In each of these instances, the conservative decision was not made. Due consideration of the consequences of not following the most conservative approach must be fully evaluated before making a final decision.

3) The team had concerns that overall, a less than questioning attitude exits within the licensee's organization. The team noted some examples where such an approach was demonstrated successfully. However, it was not apparent that this attitude was sufficiently present at all levels within the organization. The NPAD organization should be at the forefront of this effort by conducting thorough and appropriately self-critical assessments of plant operations.

During the inspection, the team observed that the organization's efforts were keenly focused on the pertinent issues, and there appeared to be a strong management influence on ensuring its expectations were communicated and executed by the staff. The team had concerns that this might not continue in the future when operations return to more of a routine nature.

7.0 Exit Meeting

A public management exit meeting was held with the licensee on July 20, 1993. Extensive local media and public interest increased the overall attendance at the meeting to approximately 100 individuals. Attachment D is a list of the meeting attendees. The team summarized the purpose of the AIT, its charter, and the findings of the inspection. This included a number of significant open issues which are discussed in Section 8.0 below. The licensee did not take issue with the noted findings, other than it was not believed that the lack of a questioning attitude was as broad a problem as it was characterized by the team. The licensee did identify as proprietary some of the documents associated with Siemens fuel that were reviewed by the team. Following the meeting, Hubert J. Miller, Deputy Regional Administrator, responded to questions from the public.

8.0 Charter Completion and Open Issues

Due to the complex technical issues and the extensive root cause analysis activities embarked upon by the licensee, the team did not complete some of the items in the charter. The team leader and Region III management agreed that the team had served its purpose, and that the outstanding issues could be tracked by NRR and regional personnel. The AIT was disbanded on July 20, 1993. Issues that remain open and will continue to be monitored by the NRC include:

1) Plans associated with inspecting the reactor core internals to evaluate potential damage to core components as well as to try and identify the presence of the missing fuel and other materials related to the fuel rod failure event. This includes the results of any recovery operations or repair activities and plans for controlling and monitoring plant operations given that all of this material may not be recovered. Based on similar situations that have occurred at other nuclear facilities, the potential impact on the public health and safety will be minimal; however, it may spread radioactive particles to various parts of the primary coolant system and may mask detection of any subsequent fuel failures.

2) Determination of the failure mechanism and the root cause of the failed fuel rod. The licensee identified a number of factors that may have contributed to this event. They also developed a root cause analysis that will progress based on the determination of the failure mechanism. The team believes a metallurgical examination may be beneficial in determining the failure mechanism and enhance the root cause analysis efforts.

3) Plans and justification for reconstituting the core. The licensee is developing a plan for reloading the core with fuel that will not be susceptible to the same failure. In particular, the NRC is interested in the

type of fuel assemblies that will be placed in the regions of the core next to the core shroud and the licensee's justification for using these assemblies.

4) The licensee has identified a number of potential causes for the stuck assembly in the UGS, building on their experiences in 1988 and 1992. The NRC intends to closely monitor the continued efforts to determine the ultimate root cause for this event and the appropriate corrective actions.

5) Effectiveness of NPAD in critically evaluating plant operations. The licensee has recognized the need to improve the staffing of NPAD and this inspection indicated that a more critical approach is needed in evaluating plant operations, such as the areas of procedural adherence and supervisory effectiveness.

6) Effectiveness of empowerment philosophy. The team noted that implementation of this management approach is incomplete and until it is fully ingrained throughout the organization, a greater level of involvement by management appears to be warranted.

<u>Palisades Fuel Failure and Fuel Handling Problems</u>
<u>Augmented Inspection Team (AIT) Charter</u>

You and your team are to perform an inspection to accomplish the following:

1. Determine and validate the sequence of events associated with the Palisades fuel handling problems that occurred on July 6, 1993, and the failed fuel assembly/lost pin event that was discovered on July 1, 1993.

2. Evaluate the failed fuel assembly/lost pin incident for the following:

   a. failure mechanism
   b. failure indications and why failure wasn't identified earlier from indications such as radiochemistry
   c. licensee's root cause analysis
   d. adequacy of licensee plans and procedures for proceeding with the inspection and removal of the fuel assembly suspected to be the source of the dropped fuel pin (I-24)
   e. adequacy of additional plans and procedures associated with other fuel assemblies, the reactor vessel, and the core internals area
   f. whether the additional fuel fragment discovered on June 29, 1993 is indicative of additional problems.
   g. licensee corrective actions and actions required pre-startup/post-startup

   Debrief concerns, questions and issues to the licensee in as complete and prompt fashion as possible so the licensee can factor these issues into investigations and corrective actions.

3. Evaluate the July 6 fuel handling event and previous events (1992 and 1988) for the following:

   a. licensee's root cause analysis
   b. procedure quality and adherence
   c. personnel performance issues
   d. attention to detail
   e. refueling work controls and practices
   f. management involvement
   g. licensee corrective actions
   h. Compare causes and corrective actions from June 15, 1993 event (control rod coupled during head lift), and 1992 and 1988 fuel handling events to determine if licensee actions should have prevented this recent occurrence.

   In evaluating this failure, strongly consider related root causes and determine if these failures are indicative of any broad weaknesses.

4. Evaluate and assess the licensee's understanding, ownership and responsibility for the Palisades reactor core assembly design.

5. Review the adequacy of the licensee's program for evaluating these events. Determine if any equipment needs to be quarantined. Oversee troubleshooting, testing and analysis of involved or quarantined equipment.

6. Interview plant personnel involved in the events to determine if personnel actions and procedural guidance were adequate.

7. Evaluate licensee managerial performance related to these events including shift supervision, management oversight and management response. Evaluate whether excessive management pressure was exerted to expedite activities between the June 15, 1993 and July 6, 1993 events.

8. For broad issues and concerns identified by the team, determine if and to what extent licensee quality assurance/verification identified similar concerns in audits and reviews of licensee operations and outage activities. Assess whether licensee QA/QV activities conducted in the recent past were capable (i.e., of adequate scope and depth) of finding such problems where they exist.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN ILLINOIS 60137

Attachment B
pg 1 of 3

JUL 0 8 1993

Docket No. 50-255
License No. DPR-20
CAL No. RIII-93-010

Consumers Power Company
ATTN: Mr. David P. Hoffman
Vice-President, Nuclear
Operations
1945 W. Parnall Road
Jackson, MI 49201

Dear Mr. Hoffman:

SUBJECT: CONFIRMATORY ACTION LETTER (CAL) RIII-93-010

This confirms the conversation on July 8, 1993, between Messrs. Hubert J. Miller, Deputy Regional Administrator, Region III, and Gerald B. Slade of your staff related to numerous problems experienced at Palisades during refueling and other previous fuel handling activities. These problems include: the failure to uncouple a control rod prior to lifting the reactor vessel head, extensive damage to a fuel rod, the inadvertent lifting of a fuel bundle during upper guide structure removal, and the unauthorized use of the override key switch on the spent fuel pool fuel handling machine. With respect to these matters, we understand that you will perform the following actions:

1. Conduct an investigation to determine the root causes of the damage to fuel bundle I-24 and the inadvertent lifting of fuel bundle SAN-8 with the upper guide structure (accounting for the similar events in 1988 and [illegible]). You will also develop appropriate corrective actions to prevent recurrence of these type of events. Documentary evidence of your investigation and corrective actions will be maintained and made available to the NRC.

2. Arrange to have an independent safety assessment to review the facts surrounding the damage to fuel bundle I-24 and the recent and previous inadvertent lifting of fuel bundles with the upper guide structure. This assessment will also determine whether the licensee's root cause analyses and corrective actions are appropriate. The assessment will be documented and made available to the NRC.

3. Conduct a prompt assessment of the recent problems experienced during refueling and fuel handling activities to determine what near-term corrective actions are necessary prior to resumption of these activities. This assessment will consider, but will not be limited to, the adequacy of management oversight and operator training and the adequacy and adherence to procedures. You will review the results of this assessment and near-term corrective actions with the NRC's Augmented Inspection Team prior to the resumption of refueling or fuel handling activities, other than actions necessary to place the reactor in a safe condition.

4. Maintain senior site or corporate managers onsite at all times to visually observe work activities in progress throughout the remainder of the current refueling outage. A senior site manager is defined as one level below the plant general manager.

5. Promptly quarantine equipment determined to be appropriate by the Augmented Inspection Team.

6. Plan to meet with Senior NRC Management prior to resumption of power operations to discuss the results of the stated assessments and the corrective actions.

None of the actions specified herein should be construed to take precedence over actions which you feel necessary to ensure plant and personnel safety.

Pursuant to Section 182 of the Atomic Energy Act, 42 U.S.C. 2232, you are required to:

1) Notify me immediately if your understanding differs from that set forth above.

2) Notify me if for any reason you cannot complete the actions within the specified schedule and advise me in writing of your modified schedule in advance of the change, and

3) Notify me in writing when you have completed the actions addressed in this Confirmatory Action Letter.

Issuance of this Confirmatory Action Letter does not preclude issuance of an order formalizing the above commitments or requiring other actions on the part of the licensee; nor does it preclude the NRC from taking enforcement action for violations of NRC requirements that may have prompted the issuance of this letter. In addition, failure to take the actions addressed in this Confirmatory Action Letter may result in enforcement action.

The responses directed by this letter are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and your response will be placed in the NRC Public Document Room.

We will gladly discuss any questions you may have concerning this matter.

Sincerely,

J Miller for

John B. Martin
Regional Administrator

cc: Michael G. Morris, Chief
Operating Officer
Gerald B. Slade, General
Manager
David W. Rogers, Safety and
Licensing Director
OC/LFDCB
Resident Inspector, RIII
James R. Padgett, Michigan
Public Service Commission
Michigan Department of
Public Health
J. M. Taylor, EDO
J. H. Sniezek, DEDR
H. L. Thompson, DEDS
T. E. Murley, NRR
J. G. Partlow, NRR
J. W. Roe, NRR
J. A. Zwolinski, NRR
E. L. Jordan, AEOD
J. Lieberman, OE
J. R. Goldberg, OGC
W. Dean, NRR
A. H. Hsia, NRR
R. J. Strasma, RIII
Big Rock Point SRI

ATTACHMENT C

PALISADES ENTRANCE ATTENDEES

JULY 8, 1993

CPCO

S. Wawro
G. Szczotka
M. Hop
V. James
B. Van Wagner
C. Ritt
H. Heavin
R. Rice
B. Gerling
K. Haas
D. Rogers
K. Osborne
G. Slade
V. Bellfuss
K. Toner
D. VandeWalle
R. Margol
G. Goralski
P. Kluskowski
T. Duffy
M. Savage
J. Haumersen
D. Smedley

US NRC

A. Hsia
W. Shafer
R. Lerch
W. Dean
A. Dunlop
J. Davis
S. Wu
D. Passehl
M. Parker

STATE GOVERNMENT OFFICIAL

R. Whale

ATTACHMENT D

PALISADES EXIT ATTENDEES

JULY 20, 1993 AT 2:00

<u>US NRC</u>

K. Salehi
J. House
A. Hsia
R. Lerch
H. Miller
W. Forney
W. Dean
C. Gill
J. Davis
A. Dunlop
R. Jones
B. McCabe
D. Passehl
W. Shafer
M. Parker
J. Strasma

<u>CPCO/CONSULTANTS</u>

D. Hoffman
G. Slade
C. Macinnis
M. Savage
E. Harbinson
G. Szczotka
D. Rogers
K. Osborne
B. Gerling
K. Haas
R. Rice
R. McCaleb
C. Ritt
T. Palmisano
J. Hanson
J. Kuemin
B. Clark
S. Armbrister
D. McBride
L. Rawson
R. Sinofeman
M. Granchi

<u>STATE GOVERNMENT OFFICIALS</u>

D. Hahn
R. Whale
J. Padgett

<u>MEDIA</u>

D. Cogswell
R. Shier
S. Lowe
D. Weinstock
D. Watts
J. Van Doren
M. Martin
R. Heibutzki
S. Oswalt

<u>PUBLIC</u>

J. Sarno
M. Roche
D. Roche
J. Stanger
A. Brown
C. Seabury
B. Householder
C. Carr
M. Stracke
K. Haffner
B. Glidden
R. O'Connor
A. O'Connor
B. Hirt
A. Hirt
J. Stewart
K. Richards
B. Trumbull
M. Jones
J. Gardner
J Gardner
V. Haldron
M. Carr
T. Cabala
B. Clark
C. Irvin
C. Runnells
B. Harris
D. Raschke
J. Goolsby
D. Rice
W. Kizs
S. Lappan
C. Schlacks
M. Roche
B. Glidden

ATTACHMENT E

SUMMARY OF FUEL ROD TESTING RESULTS

Visual Examination of I-24

The visual inspection of I-24 was started on July 13, 1993, by lowering the fuel assembly from the spent fuel pool hoist box onto the fuel elevator and observing the fuel assembly with a submerged TV camera. This process resulted in an inspection that started from the bottom of the fuel assembly and therefore, the bottom spacer is designated spacer number 1. The first visual inspection involved looking at an undamaged corner of fuel assembly I-24. Spacers 1 through 7 were examined with no problems noted. On July 14, the C-B corner of fuel assembly I-24 was examined (this was the corner with the missing fuel rod). Figure 6 summarizes the examination results from sides B and C of assembly I-24. Spacer 1 had damage to the top of the corner of the spacer and an adjacent fuel rod (A-14) showed fretting damage. There was either debris or a wear mark between the adjacent fuel rods. Spacer 2 was bowed out and may have been split with little or no material missing. Spacers 3 and 4 had considerable material missing and a rub mark on an adjacent rod in spacer 4. Spacer 5 was cracked, but no material was missing. Spacers 6, 7, 8, and 9 had the entire corner of the spacer missing. Spacers 6 and 7 had debris stuck between the adjacent fuel rods and spacer 7 had wear marks on the fuel rods adjacent to the missing fuel rod. Spacer 8 had the lantern spring pushed up and spacer 9 had the lantern spring missing. Spacer 10 had wear at the corner but was intact. There was a large fuel plume evident starting just below spacer 10 and covering the corner of spacer 10. The lantern spring on spacer 10 was displaced upwards. The damage to the bottom five spacers appeared to have been caused by a different mechanism than the mechanism that caused the damage to the top five spacers. The licensee postulated that the fuel plume was indicative of a fuel failure at an earlier time.

Examination of Fuel Assembly I-24

Fuel rod locations are shown in Figure 5. The fuel rod locations are identified starting with the upper left corner being A01, the upper right corner being S01, the lower left being A15 and the lower right being S15. The missing fuel rod from I-24 was from the S15 location. Figure 1 of this attachment shows the fuel rod locations that were tested and the results of the eddy current testing (ECT). The focus was on the corner locations and the fuel rods surrounding the corners. The licensee planned to examine the fuel rods adjacent to the missing fuel rod, but decided not to due to concerns about being able to return the fuel rods to these locations due to the severe damage to the spacers at these locations. The ECT showed wear indications on rod A08 that were attributed to the plenum spring. Fuel rod A15 had 14 or more wear indications from 0.007 to 0.009 inches deep. Fuel rod S01 had 10 or more wear indications 0.008 to 0.012 inches deep. Fuel rod S02 had several minor indications. All of the corner rods that showed wear indications had been located next to a corner location of the shroud for at least one cycle of operation. Corner rod A01 had never been in a corner location of the shroud and showed no wear. All of the wear indications are on the top 1/3 of the fuel rod.

Examination of Fuel Assembly I-21

Figure 2 of this attachment shows the fuel rod locations in fuel assembly I-21 that were examined and the ECT wear indications that were detected. Once again, the focus was on the rods in the vicinity of the corner. Of note was that the rods in the corners that had been adjacent to the corner of the shroud at some point all showed wear probably caused by fretting at the spacer grid locations. The wear indications range from 0.003 to 0.012 inches deep with fuel rod S02 having the most extensive fretting of any of the I-21 fuel rods. None of the remaining fuel rods examined for fuel assembly I-21 showed any wear. Corner rod A15 showed no wear. This rod had never been located at a corner of the shroud. The remaining three corner rods showed wear indications and had been located at a corner of the shroud for one full cycle. In addition, two fuel rods adjacent to the corner rods were also loose. There were no indications of cracks or holes in the cladding on fuel assembly I-21. If the fuel rods that show fretting were replaced, it would be likely that the replacement fuel rods would also be loose and suffer from fretting.

Based on the results of the I-21 fuel assembly, the licensee decided to examine additional fuel assemblies. Fuel assemblies I-48 and J-21 were selected for examination. Fuel assembly I-48 was in core location B19 during cycle 9. Fuel assembly J-21 was a proposed replacement fuel assembly.

Examination of Fuel Assembly I-48

Figure 3 of this attachment shows the fuel rod locations of assembly I-48 that were examined and summarizes the results. More of the center rods were examined to see if problems existed in other than the corner locations. Indications of wear approximately 0.006 inches deep existed on three of the four corner fuel rods at several of the spacer grid locations. This indicates that the rods were probably loose. There were no other indications. It is possible that some of the fuel rods adjacent to the corner rods are also loose as was observed on fuel rods I-24 and I-21. Corner rod A15 showed no indications and had never been located at a corner of the shroud. The other three corner rods showed wear indications and had been located next to a corner of the shroud for at least one cycle.

Examination of Fuel Assembly J-21

Rods in all four corners, three peripheral rods and four interior rods were examined as part of the assessment of assembly J-21. The only fuel rod that showed any wear indication was an interior rod, C03 with one minor indication 0.004 inches deep. None of the corner rods showed any evidence of wear. This fuel assembly had not been located near a corner shroud position during any of the previous cycles.

Examination of Fuel Assembly K-31

All four corner rods, four peripheral and four interior rods were examined as part of the assessment of assembly K-31. A minor wear indication was observed on corner rod S15 that was <0.002 inches deep. This fuel assembly had not been located near a corner shroud position during any of the previous cycles, but was adjacent to I-24 during the last cycle.

Examination of Fuel Assembly H-31

Corner rods A1 and S1 and three adjacent rods were examined as part of the assessment of assembly H-31. All of fuel rods examined had wear indications ranging from two to eight indications per fuel rod and 0.004 to 0.008 inches deep. This fuel assembly was located at a corner location (X19) during cycle 9 and had been in the core for a total of four cycles.

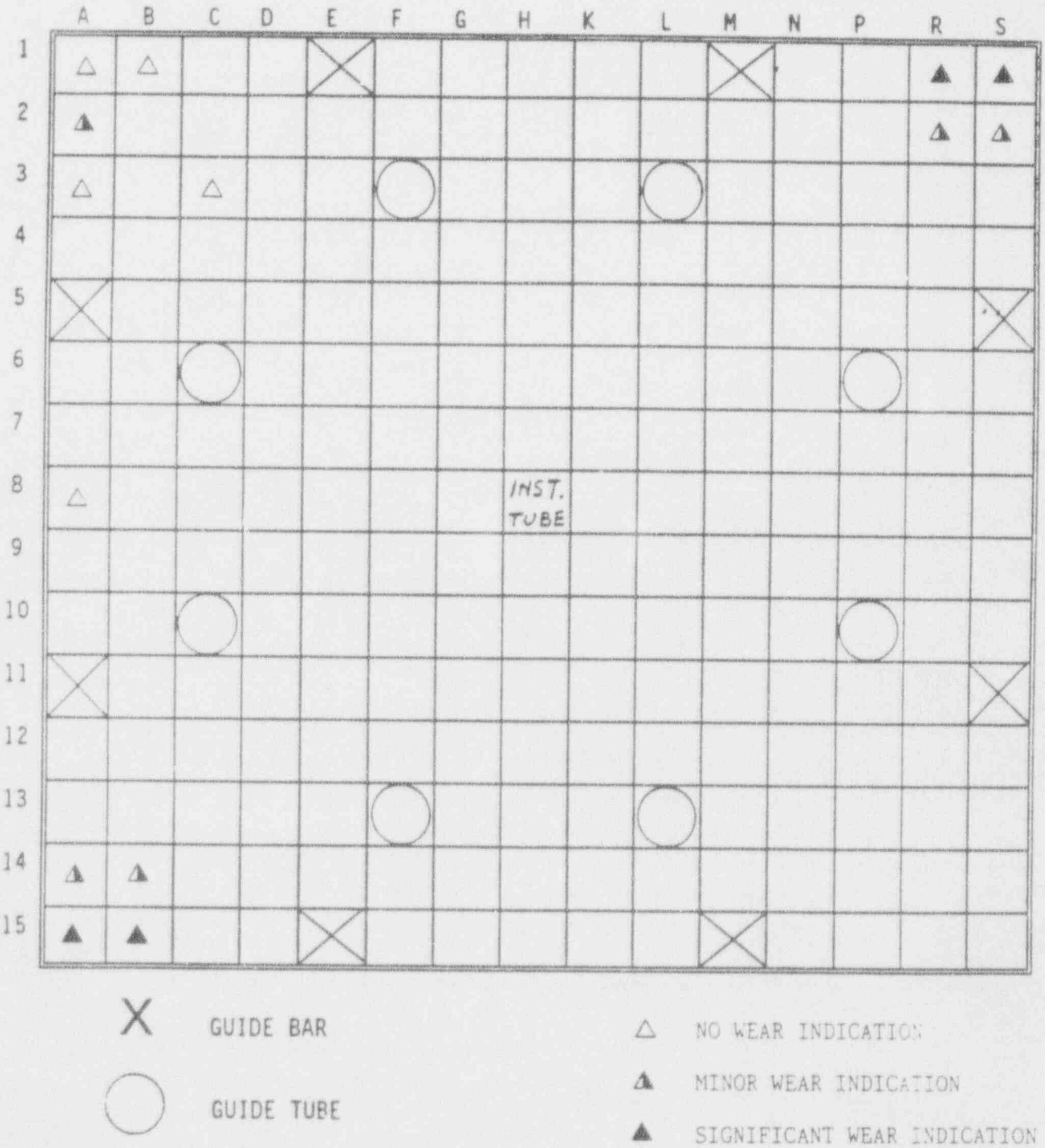
Examination of Fuel Assembly L-24

All four corner rods, four peripheral and four interior rods were examined as part of the assessment of assembly L-24. No significant indications were observed during the ECT. This fuel assembly had been adjacent to I-24 during the previous cycle in position B-18, next to the shroud and had been in the core for a total of three cycles.

ATTACHMENT E

FIGURE 1 ASSEMBLY I-024 ECT RESULTS

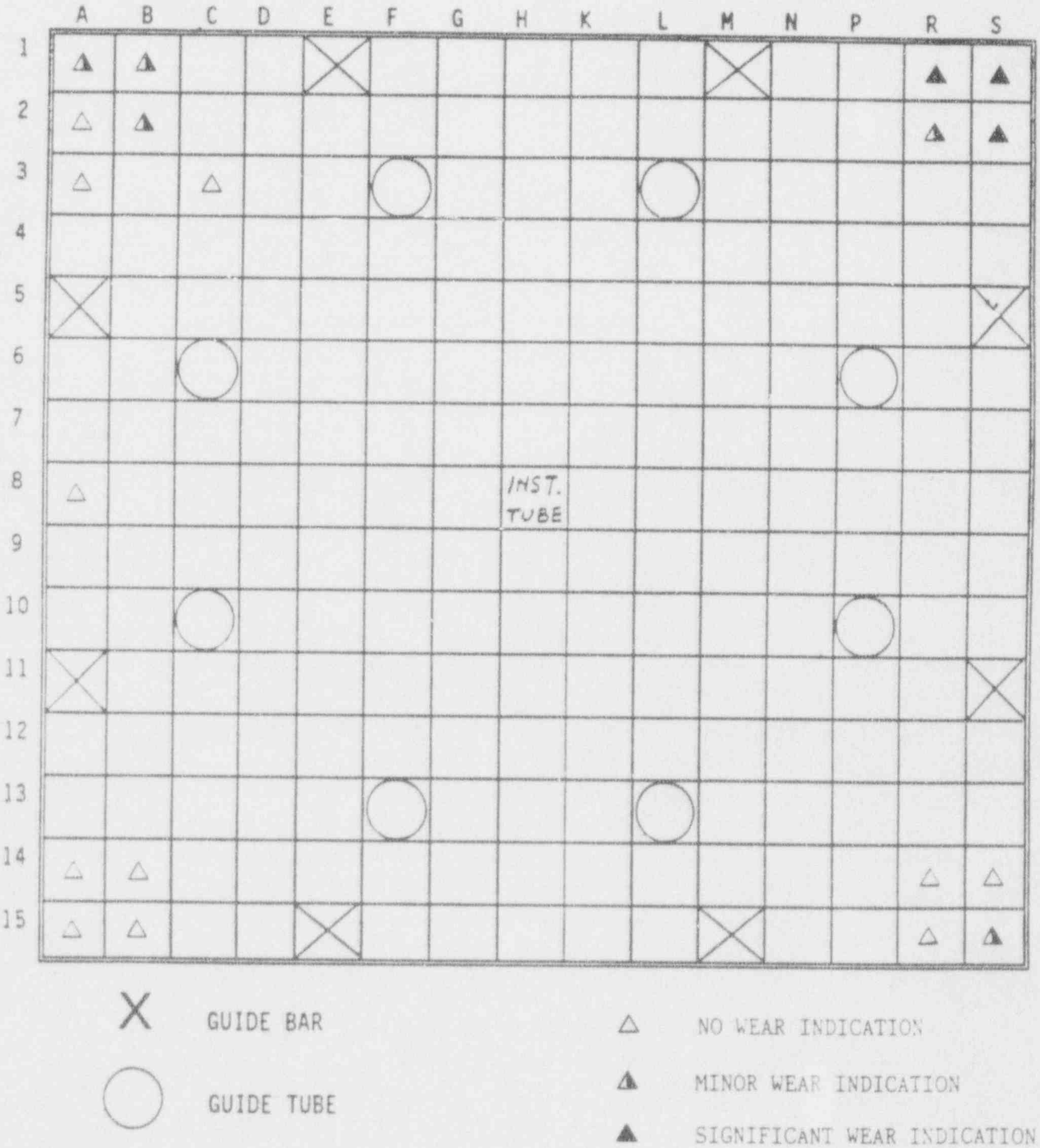
PALISADES FUEL ASSEMBLY MAP



ATTACHMENT E

FIGURE 2 ASSEMBLY I-021 ECT RESULTS

PALISADES FUEL ASSEMBLY MAP



ATTACHMENT E

FIGURE 3 ASSEMBLY I-048 ECT RESULTS

PALISADES FUEL ASSEMBLY MAP

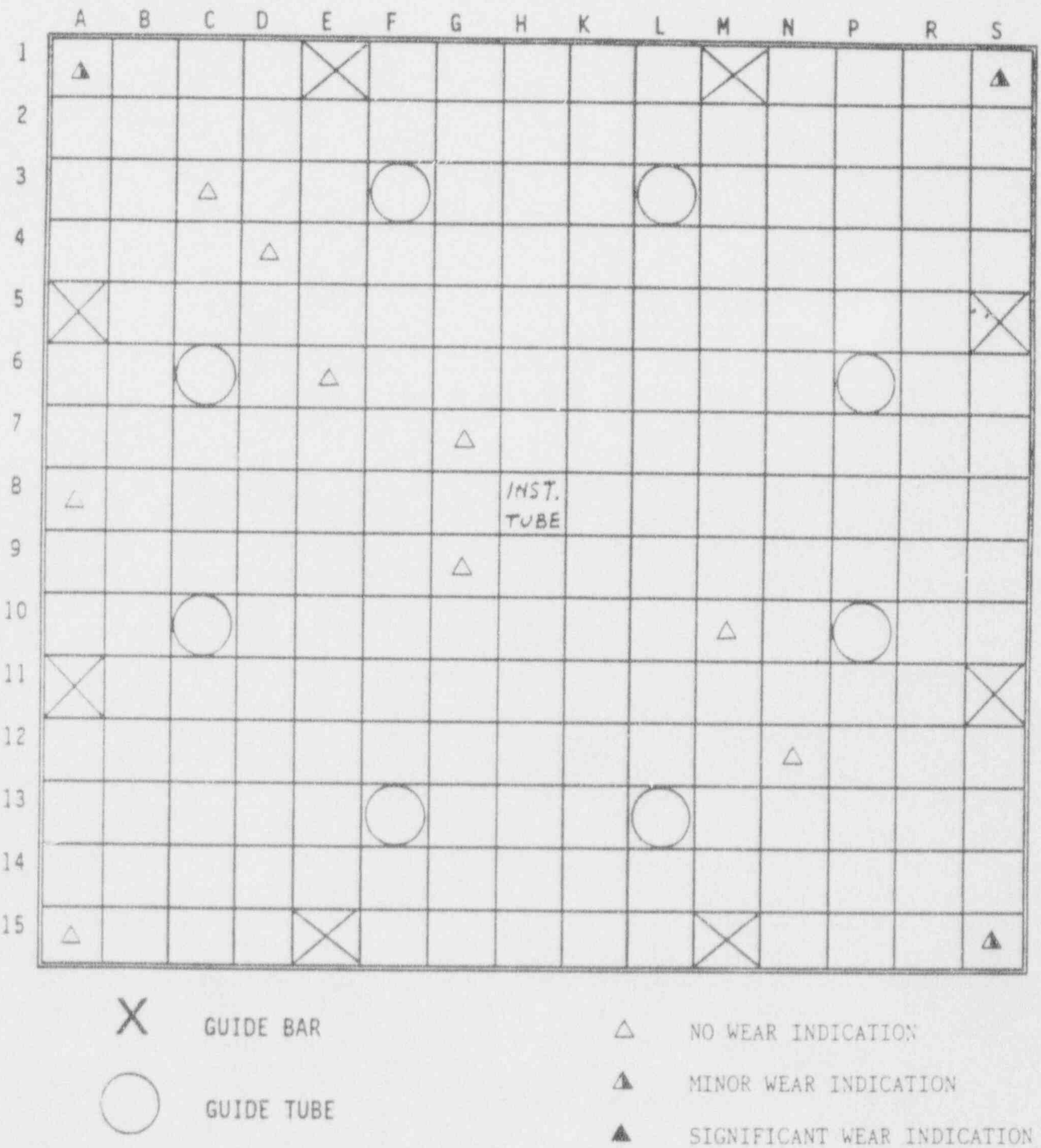
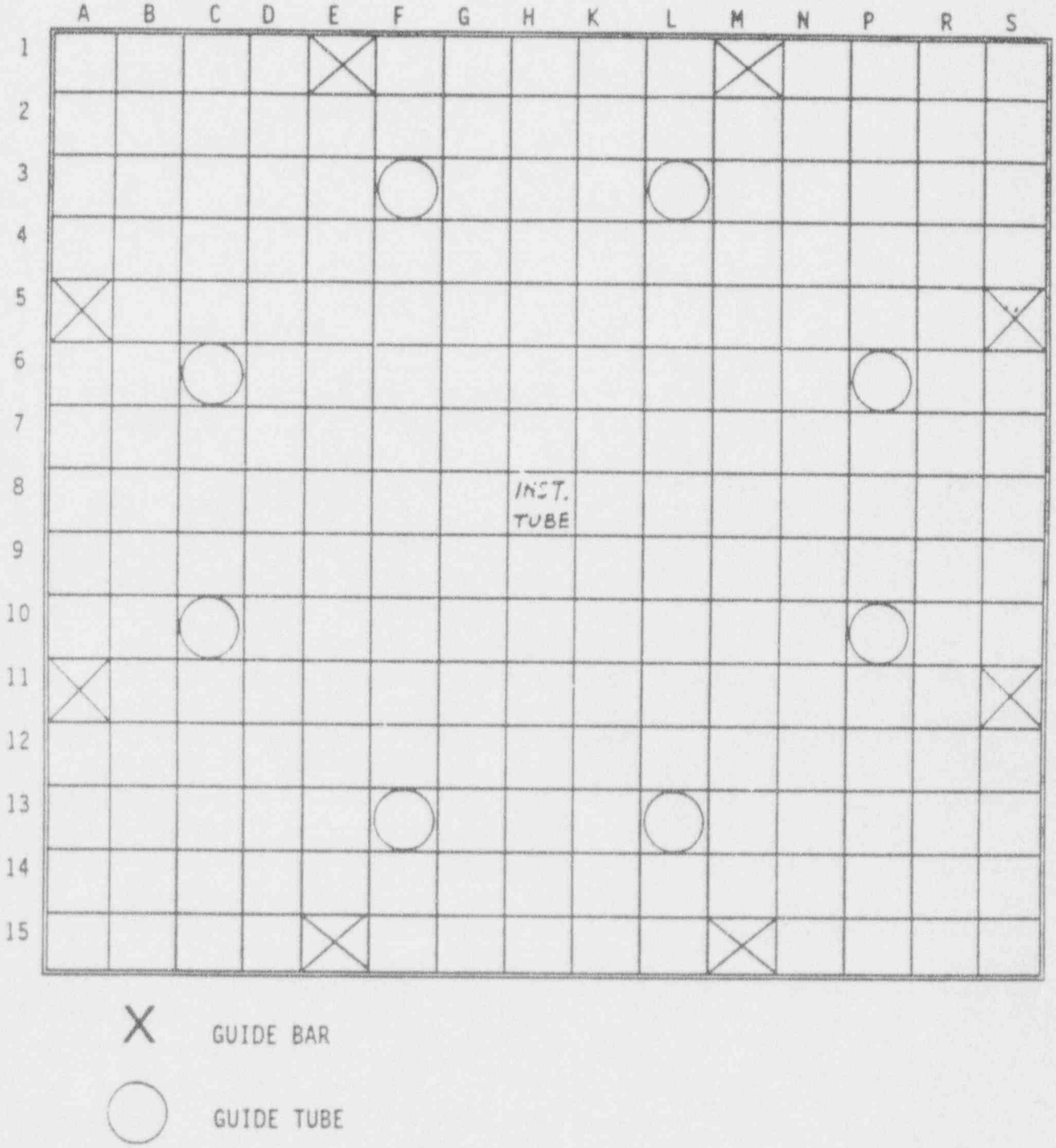
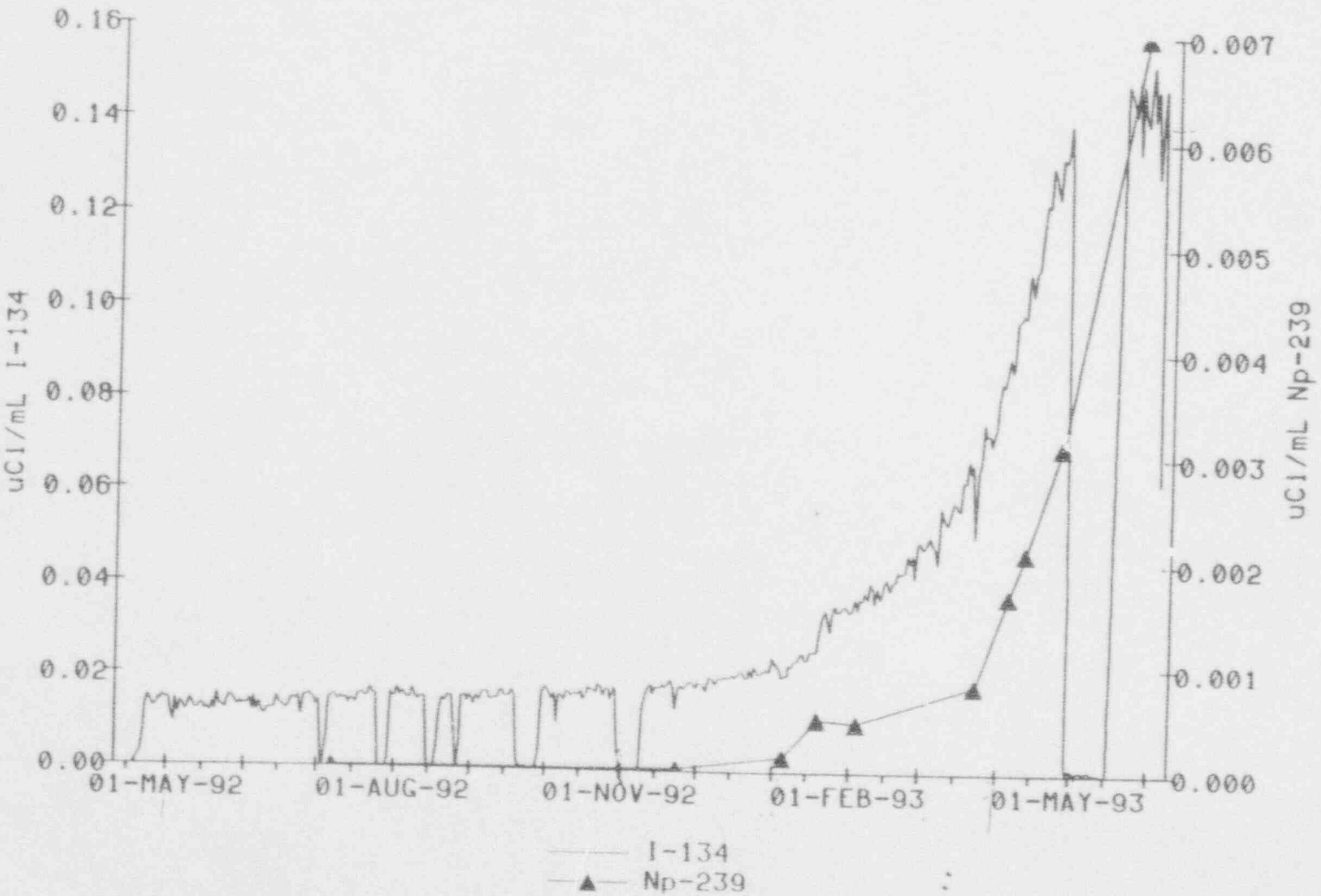


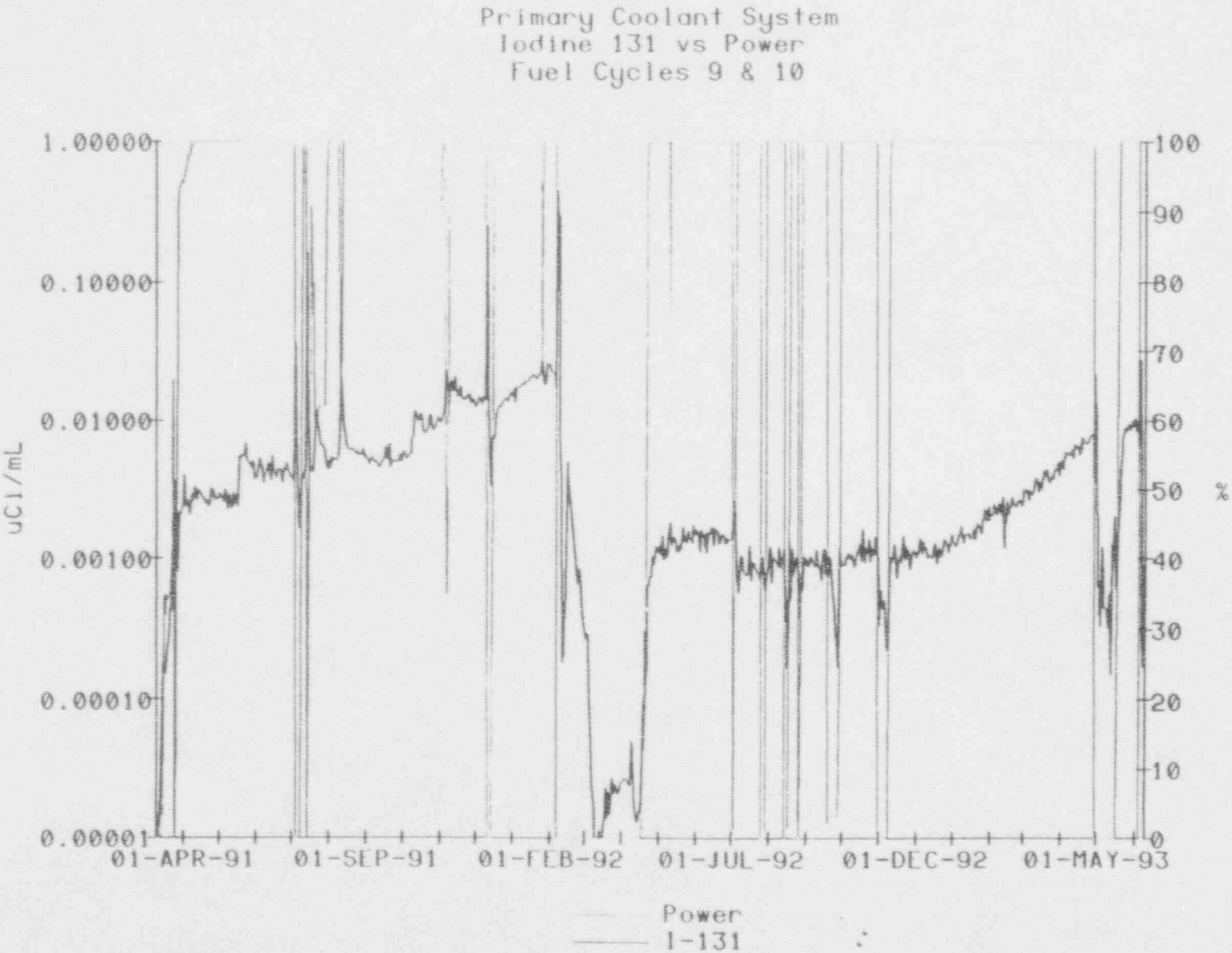
FIGURE 5

PALISADES FUEL ASSEMBLY MAP



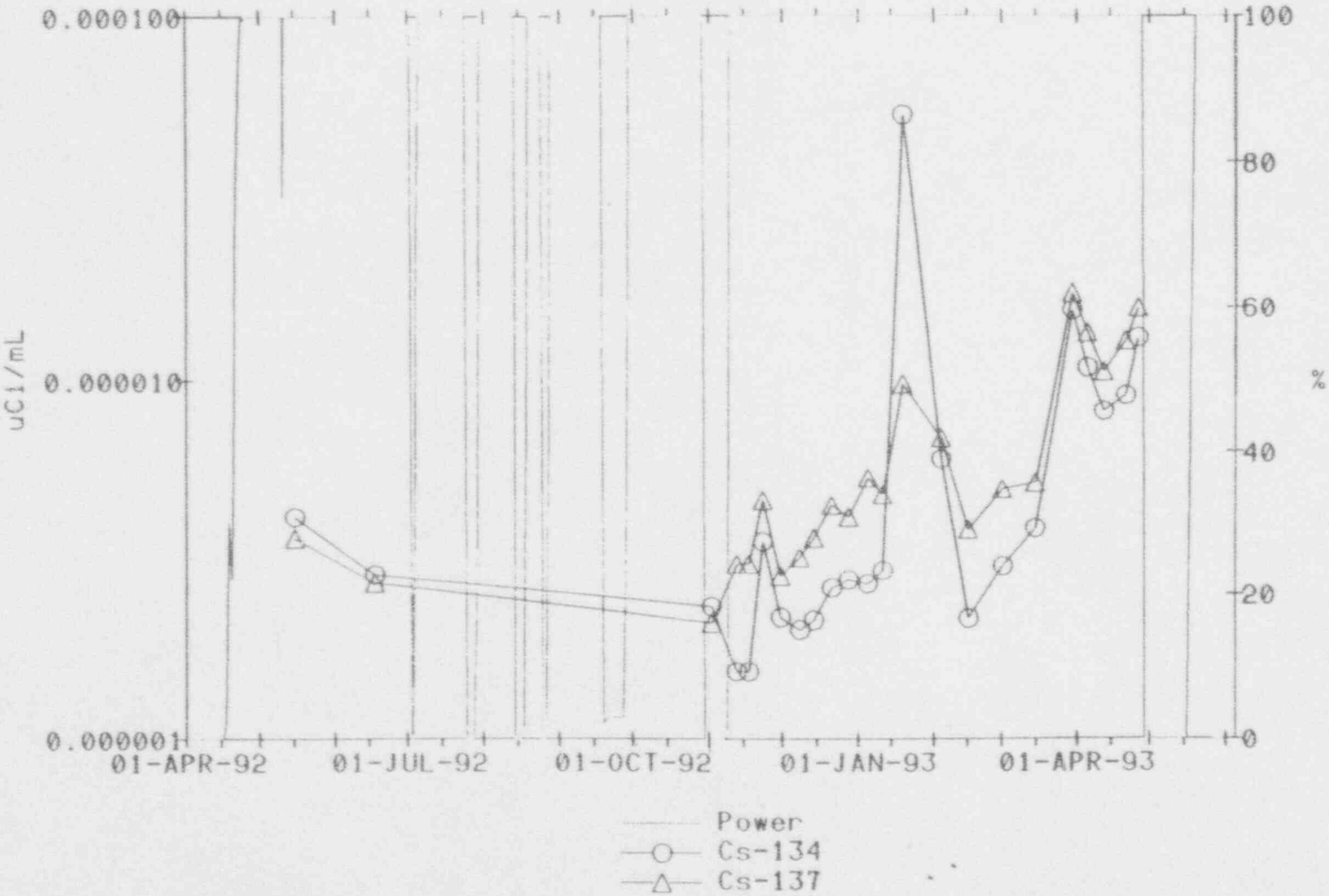
Primary Coolant System
Neptunium 239 vs Iodine 134
Fuel Cycle 10





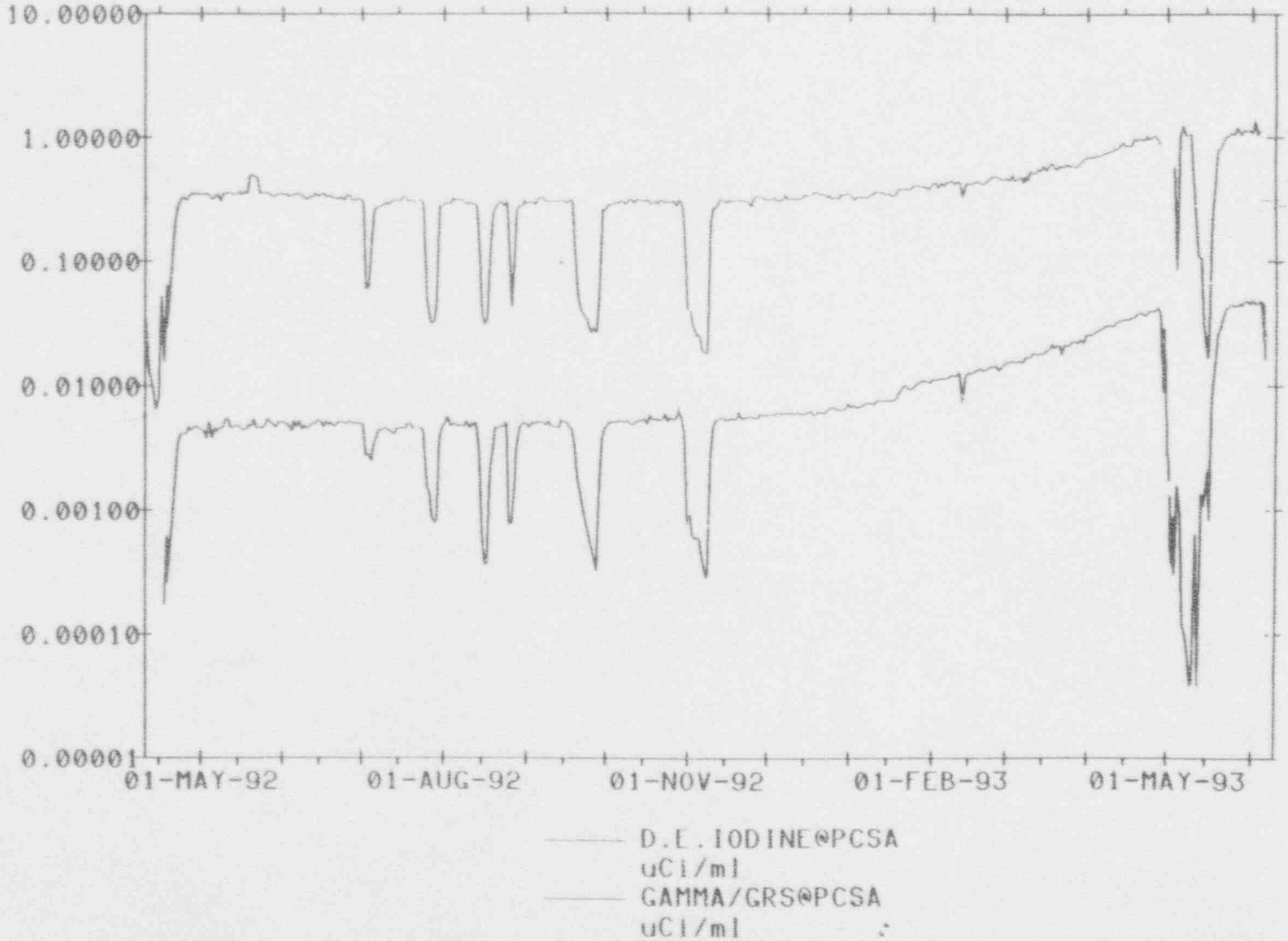
Attachment F
pg 2 of 4

Primary Coolant System
Cesium vs Power
Fuel Cycle 10



Attachment F
pg 3 of 4

Dose-Equivalent Iodine vs Gross Gamma



Attachment F
Pg 4 of 4

ATTACHMENT G

DETAILED SEQUENCE OF EVENTS FOR THE STUCK FUEL ASSEMBLY EVENT

DATE/TIME (EDT)	EVENT
7/6/93	
08:15 p.m.	Upper Guide Structure (UGS) lift evolution pre-job briefing.
09:50 p.m.	Verified that all Technical Specifications (TS) and administrative requirements were satisfied for UGS lift.
10:30 p.m.	Westinghouse contractors commenced the UGS lift. Shortly after that, the Senior Reactor Operator (SRO) inside containment informed the control room that the UGS had been raised to 6 inches. The load cell on the lift rig indicated a load of 62,800 pounds (this was higher than the expected UGS load of approximately 56,860 pounds and exceeded the limit of 62,000 pounds set by Procedure No. RVI-M-1, Rev. 16). A discussion was held between the SRO, the Shift Manager, the System Engineer, and the Westinghouse Refueling Service Supervisor. They decided to raise the UGS to 3 feet above its normal position as allowed by Procedure No. RVI-M-1, Rev. 16.
10:38 p.m.	The UGS was lifted to 3 feet above its normal position to facilitate video camera inspection. Commenced camera inspection of the under side of the UGS for possible stuck fuel assembly.
10:54 p.m.	The SRO informed the control room that assembly SAN-8 in core location Z-11 was stuck to the UGS at the 3 feet elevation. The UGS movement was suspended and the camera inspection continued. Operations staff declared an Unusual Event according to procedure and made appropriate notifications. With the exception of the crew performing the camera inspection, all non-essential personnel were evacuated from the containment.
10:57 p.m.	Operations initiated containment isolation.
11:36 p.m.	Containment integrity was established.

7/7/93

12:30 a.m.	Camera inspection was completed and it confirmed that no other assembly was attached to the UGS. All personnel were evacuated from the containment.
03:00 p.m.	Conducted pre-job briefing for recovering assembly SAN-8 from the UGS.
04:37 p.m.	Operations authorized the recovery of SAN-8 using Procedure No. FHSO-18, Recovery of Bundle SAN-8.
07:55 p.m.	Assembly SAN-8 was secured by 2 jib crane hooks with chainfalls while the assembly was still attached to the UGS. The Westinghouse crew attempted to free the stuck assembly, by alternately adjusting the tensions on the two chainfalls. This attempt was not successful.
08:48 p.m.	The equipment hatch was opened to bring in the reconfigured slide hammer in order to apply force to the assembly upper tie plate to release the assembly from the UGS fuel alignment pins.
09:20 p.m.	The SRO informed the control room that the slide hammer was too heavy to be used.
10:30 p.m.	A meeting was held between the licensee's system engineering and operations groups, and the Westinghouse staff. It was concluded that a new slide hammer must be fabricated.

7/8/93

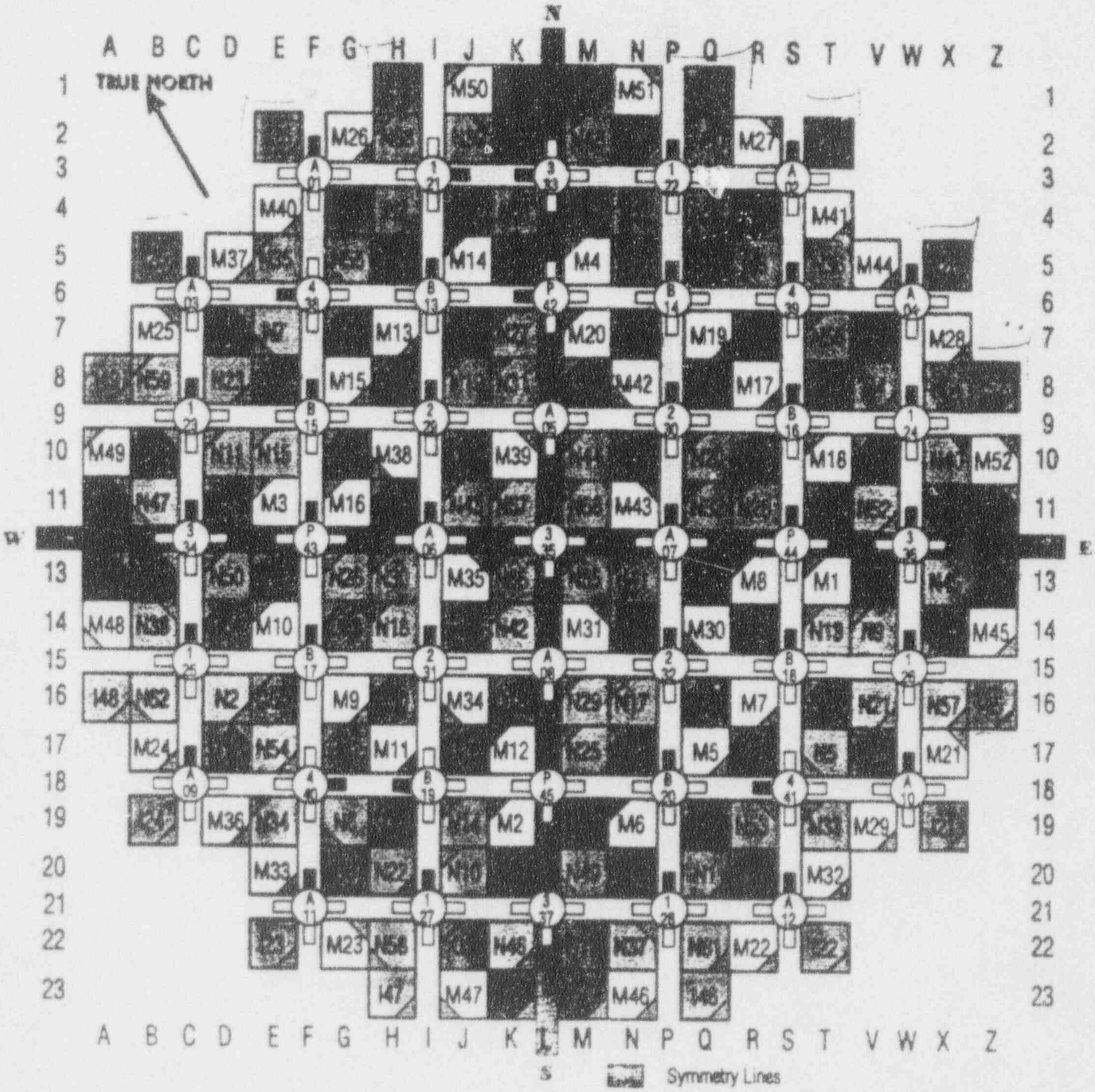
08:00 a.m.	Mockup training was conducted for use of a new replacement slide hammer.
10:00 a.m.	Conducted a pre-job briefing for the second attempt to free the assembly from the UGS (using the new slide hammer).
11:15 a.m.	The containment equipment hatch was opened to receive the new slide hammer.
12:30 p.m.	The new slide hammer was pre-staged inside containment. The equipment hatch was closed. Operations authorized using Procedure No. FHSO-18, Rev. 1, Recovery of Bundle SAN-8.

7/8/93(cont'd)

03:20 p.m.	After applying force from the new slide hammer several times to the upper tie plate, assembly SAN-8 was freed from the UGS fuel alignment pins. The assembly remained suspended from the jib crane hooks.
04:01 p.m.	Assembly SAN-8 was carefully lowered into core location Z-11 on the core support plate and the jib crane hooks were detached and removed.
04:50 p.m.	Exited from the Unusual Event and completed all required notifications.
08:15 p.m.	Conducted a pre-job briefing for movement of the UGS to its storage pads.
10:28 p.m.	The UGS was placed on its storage pads.

Figure 1
Initial

CYCLE 11 CORE PLAN
PALISADES NUCLEAR PLANT



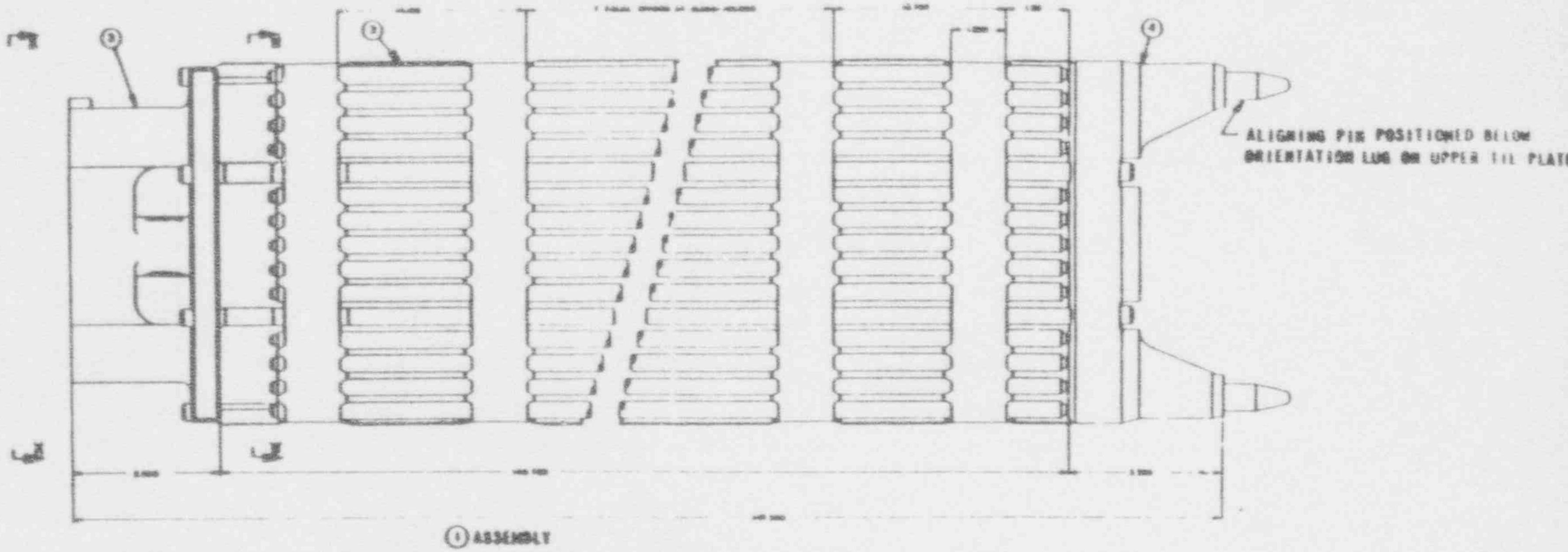
Batch O: New Fuel   Batch N: Once Burnt   SAN: Once Burnt   Batch M: Twice Burnt   Batch I: Five Times Burnt

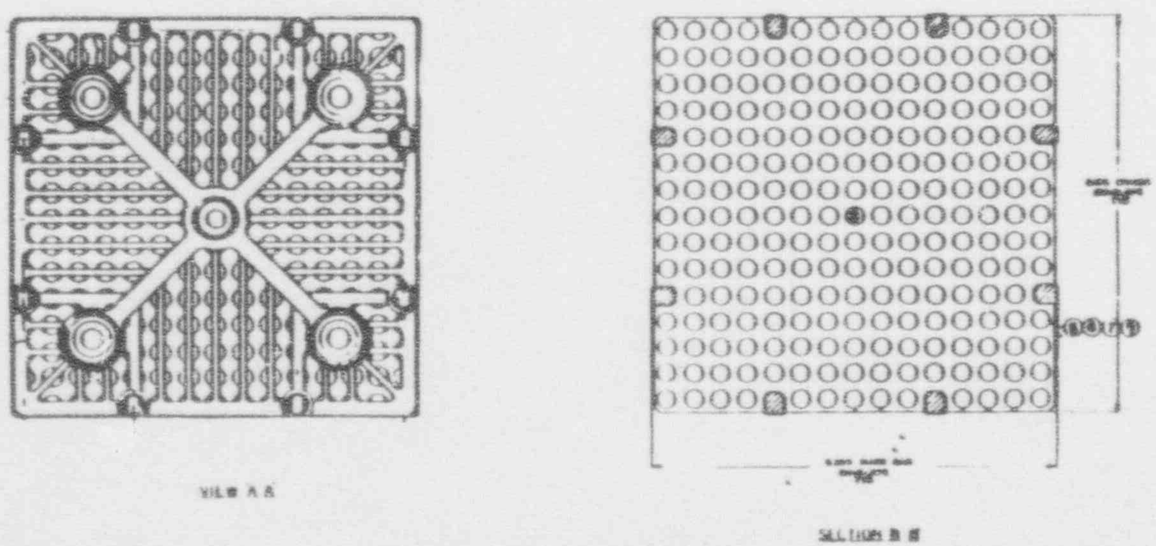
Batch I assemblies contain Halfnium clusters
SAN assemblies contain Stainless Steel pins

Cycle 11 Core Plan uses 1/4 core rotational symmetry

COREMAP.CDR
Rev 5/28/93

Figure 2

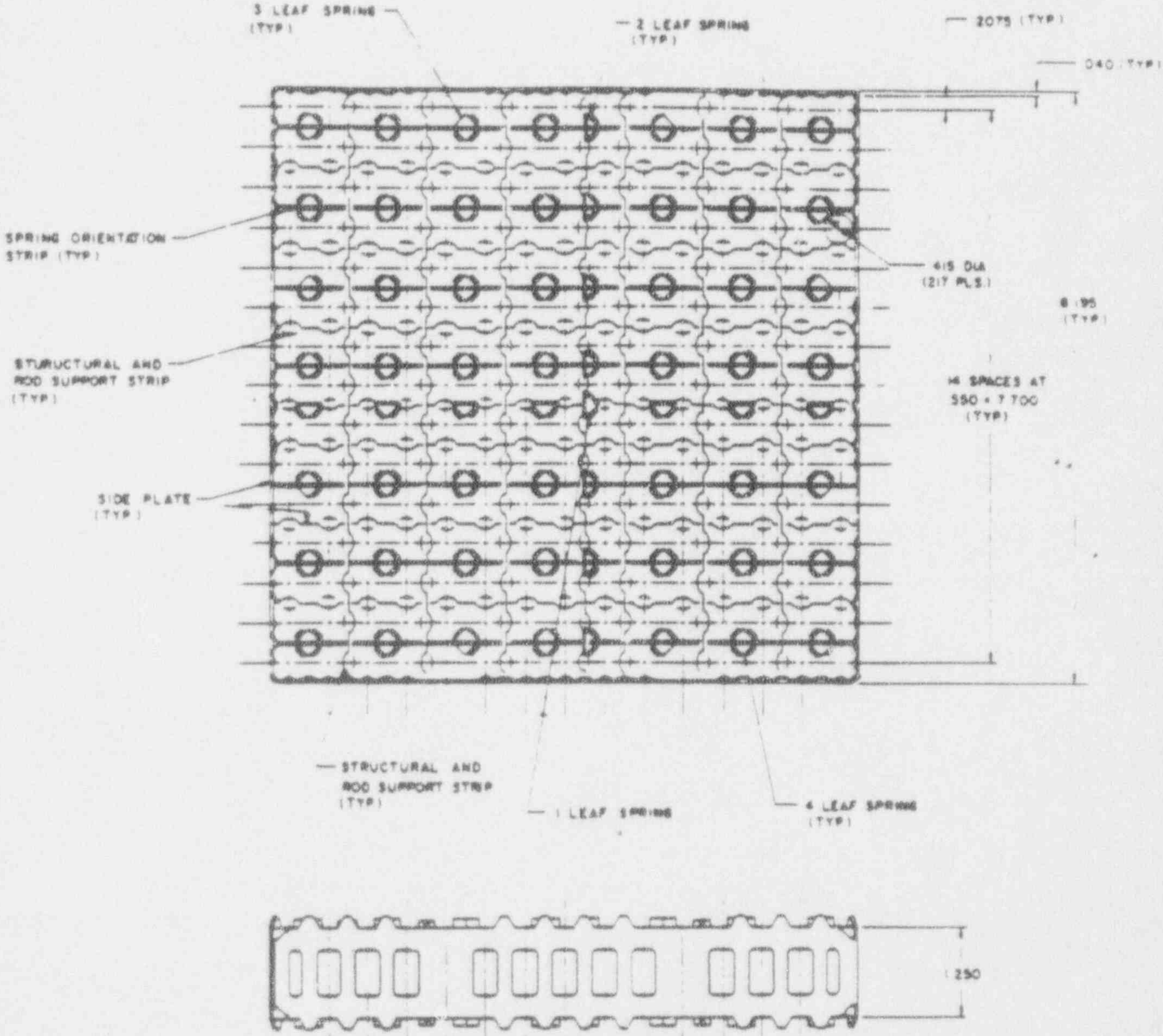




GENERAL NOTES

1. DIMENSIONS AND TOLERANCES ARE IN ACCORDANCE WITH ANSI Y14.5.1966
2. CAGE ASS'Y
3. UPPER TIE PLATE ASS'Y
4. LOWER TIE PLATE ASS'Y
5. STANDARD ROD (HIGH ENRICHMENT)
6. STANDARD ROD (MED ENRICHMENT)
7. STANDARD ROD (LOW ENRICHMENT)
8. INSTRUMENT TUBE ASS'Y
9. POISON ROD

CONSUMERS POWER COMPANY PALISADES PLANT FSAR UPDATE
FUEL BUNDLE ASSEMBLY TYPICAL FOR BATCHES H, I, J
FIGURE NO 3-22



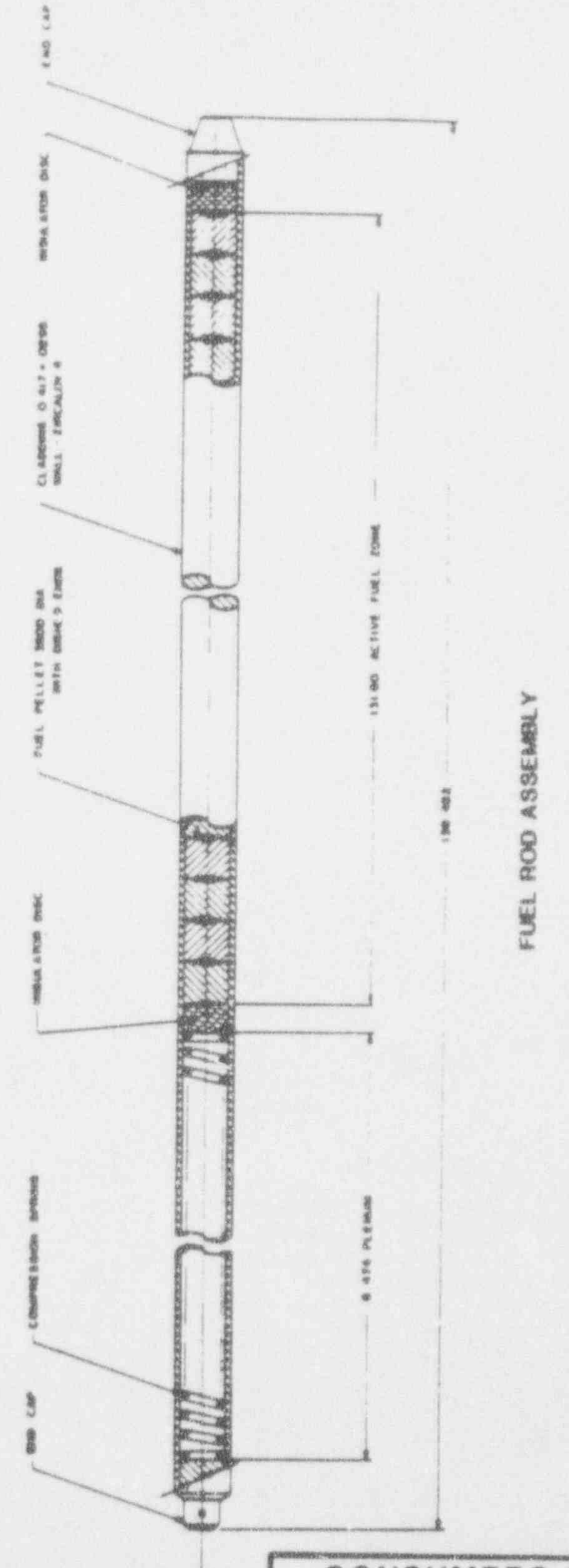
SPACER ASSEMBLY

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

EXXON NUCLEAR
SPACER ASSEMBLY

TYPICAL FOR BATCH E - L

FIGURE NO 3-23



CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

EXXON NUCLEAR
FUEL ROD ASSEMBLY

TYPICAL FOR BATCHES H, I, J

FIGURE NO 3-24

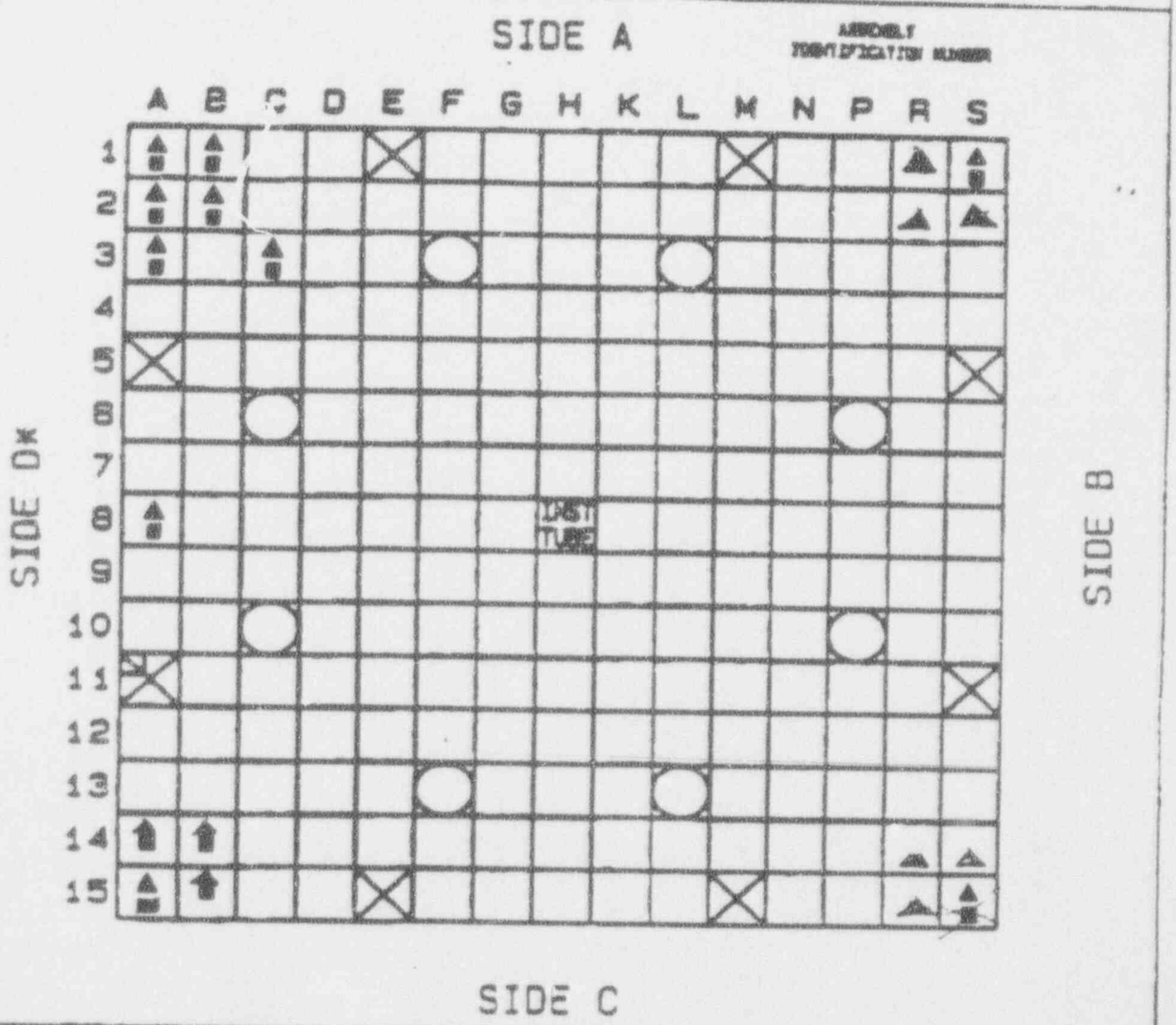
CONVENTION FOR IDENTIFYING SUBJECT FUEL RODS FOR INSPECTION

NOTE: Alternate rods may be selected at R&SAE Representative's authorization.

ADVANCED NUCLEAR FUELS CORP.

FUEL ASSEMBLY MAP

REACTOR: PALISADES		ASSEMBLY VIEW: TOP
ASSEMBLY NUMBER:	MATRIX: 15 X 15	REACTOR TYPE: PWR



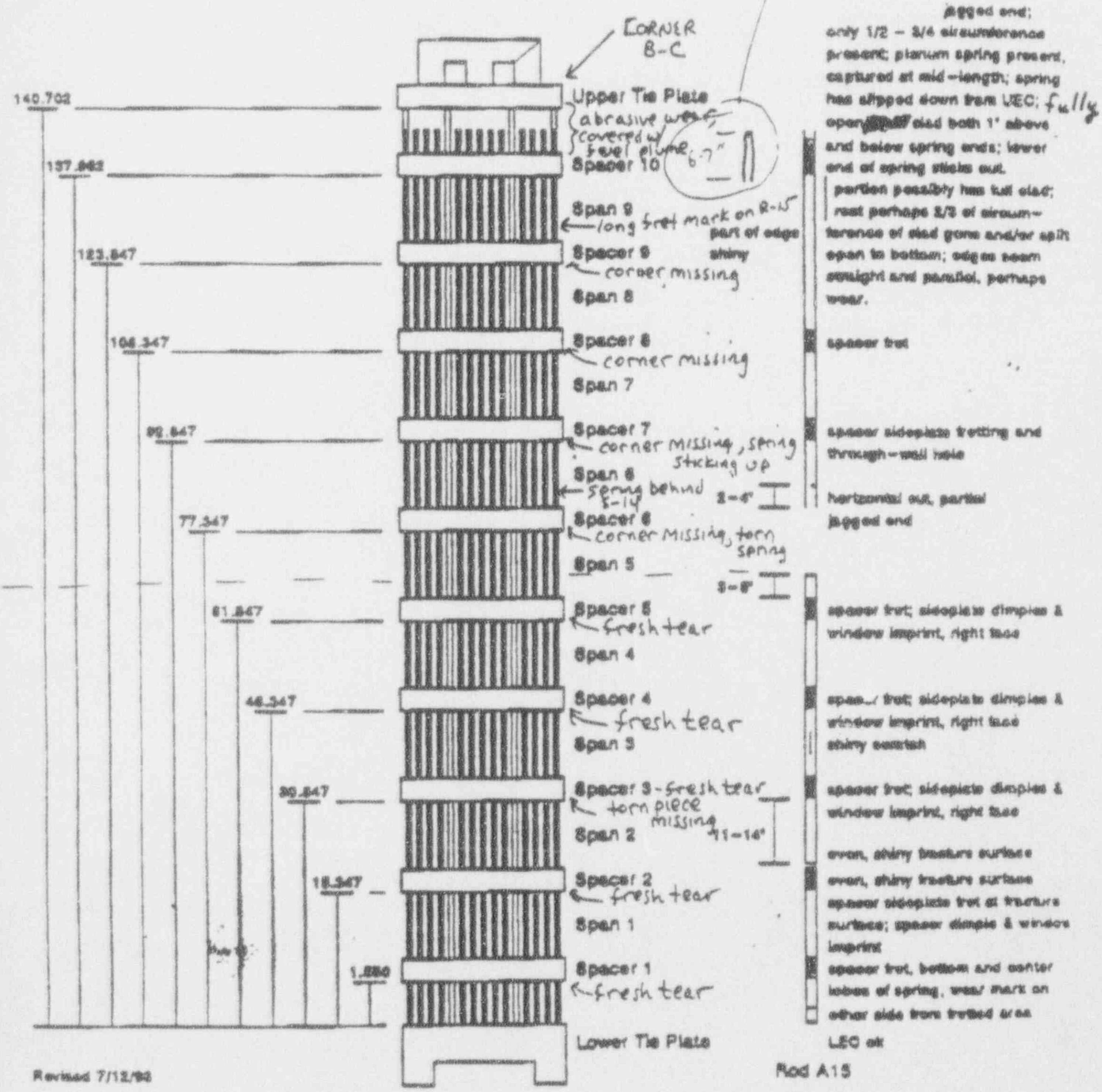
GENERAL NOTES: GUIDE BARS
Rod Withdrawal Force, Rod Visuals, Rod Eddy Current
Rod Withdrawal Force, Rod Visuals, Rod Eddy Current
Guide Tubes

MAP NUMBER	REFERENCE DRAWING
ANF-FP-15T-UD08 REV. 1	FUEL BUNDLE ASSEMBLY: XN-NF-302, 971

Figure 6

Piece located 7-13-93, placed into storage basket

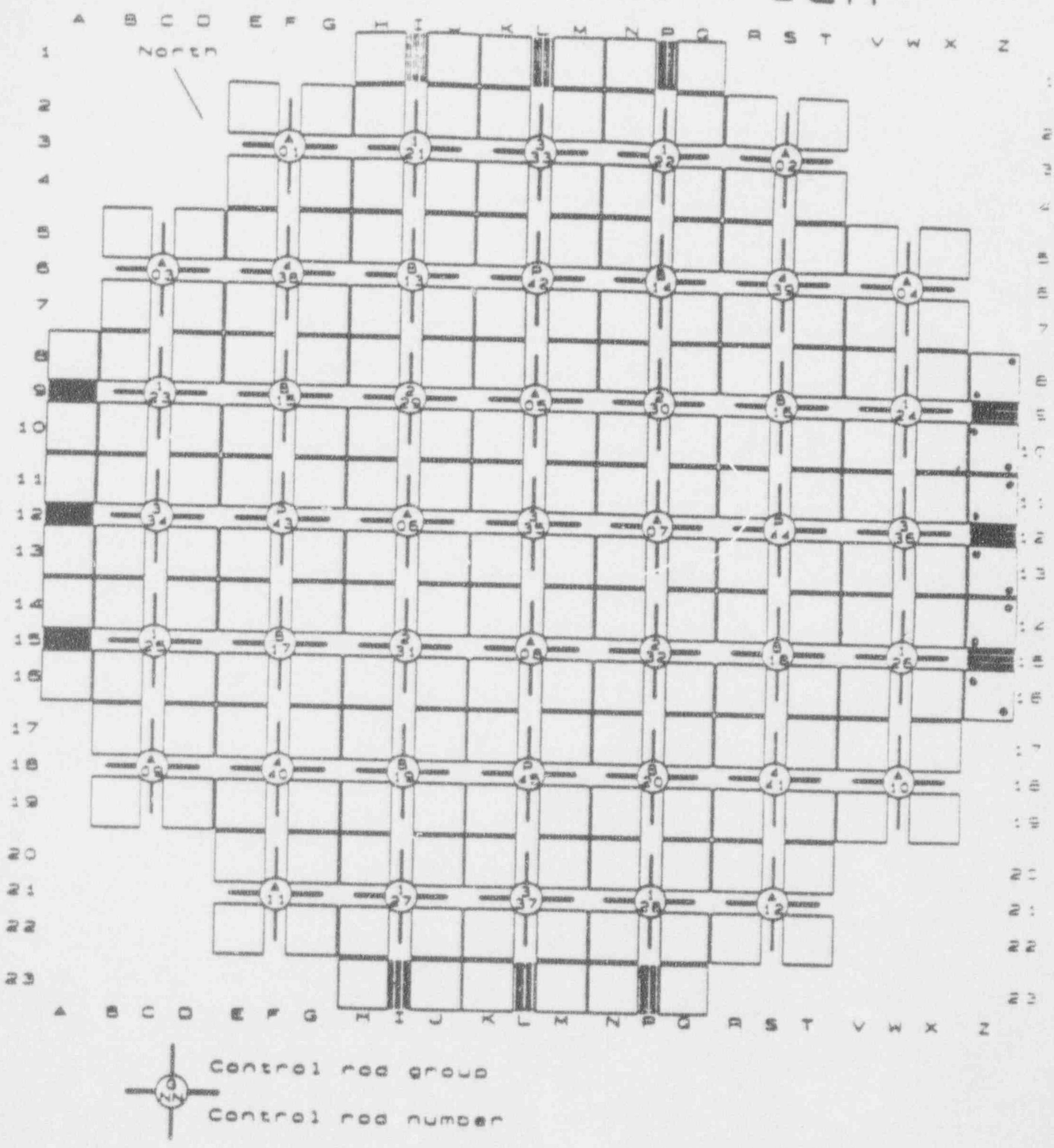
PALISADES ASSEMBLY I-024



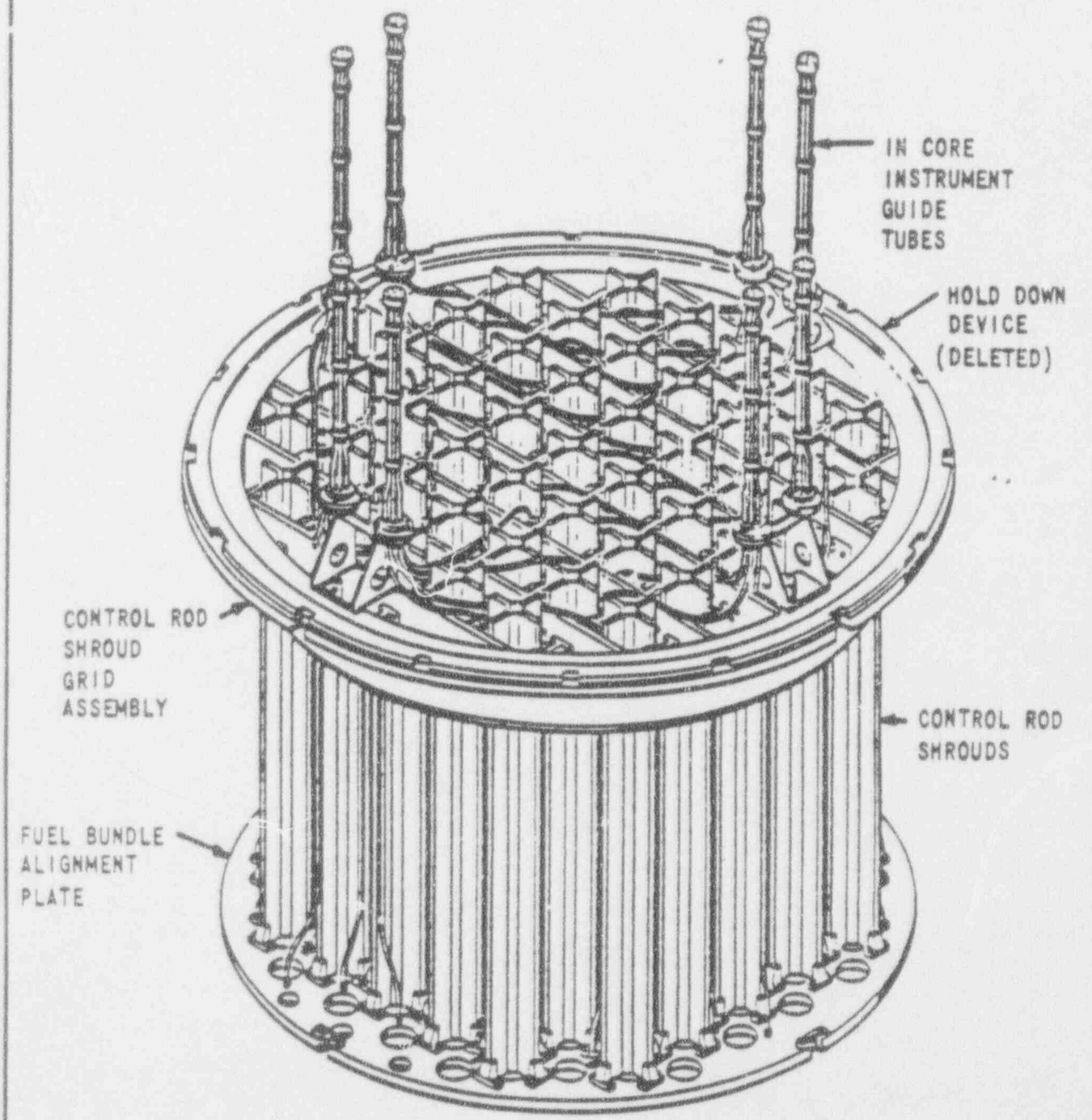
RAPIFAX NO. F5-653
PAGE 1 OF 2
ATTN:

Figure 7

Palisades Nuclear Plant Reactor Core Plan



Form from EGAD 13 revision 0



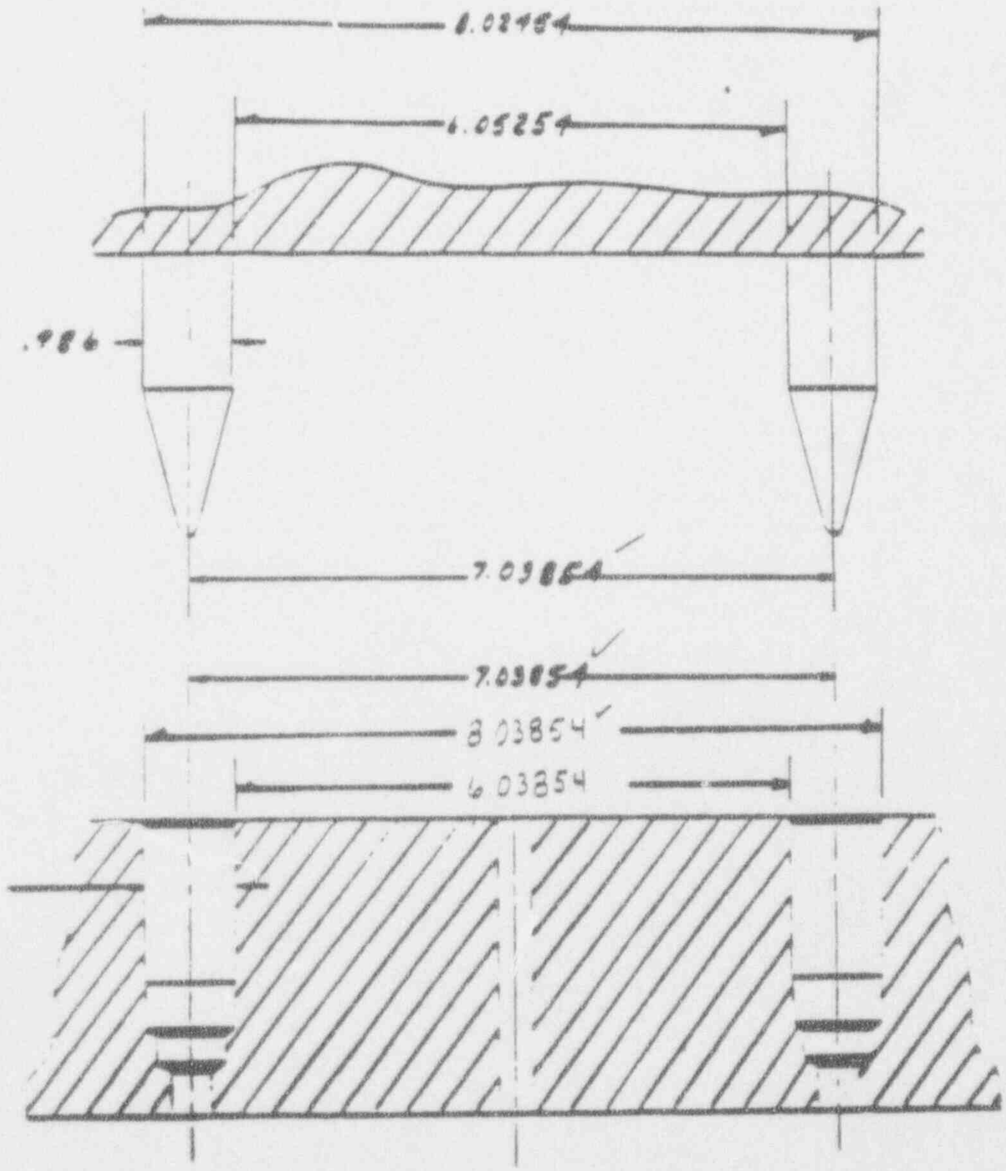
CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

UPPER GUIDE STRUCTURE ASSEMBLY

FIGURE NO 3-19

REVISION NO 0

Reference/Comment



Configuration : Nominal Alignment Pin Tolerances
Nominal Tie Plate Tolerances

Clearance : 8.03854 - 8.02454 = $\boxed{0.014}$ = 6.05254 - 6.03854

This represents a minimum clearance for the nominal configuration. i.e. The alignment pin is shifted all the way to one side of the tie plate mating hole.

Reference/Comment

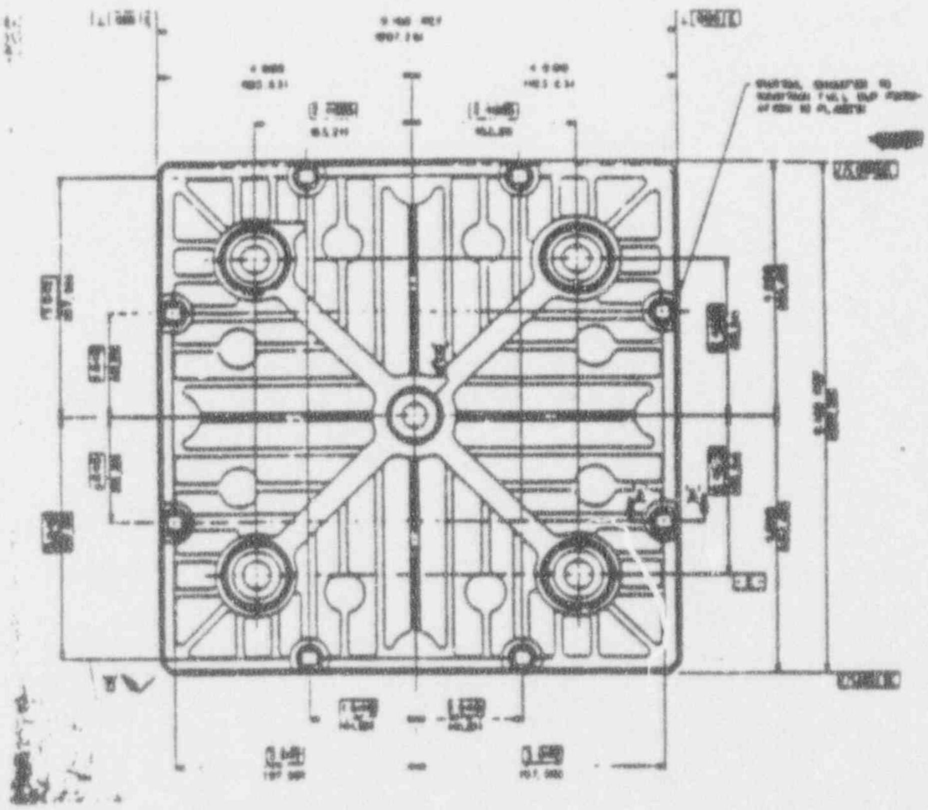
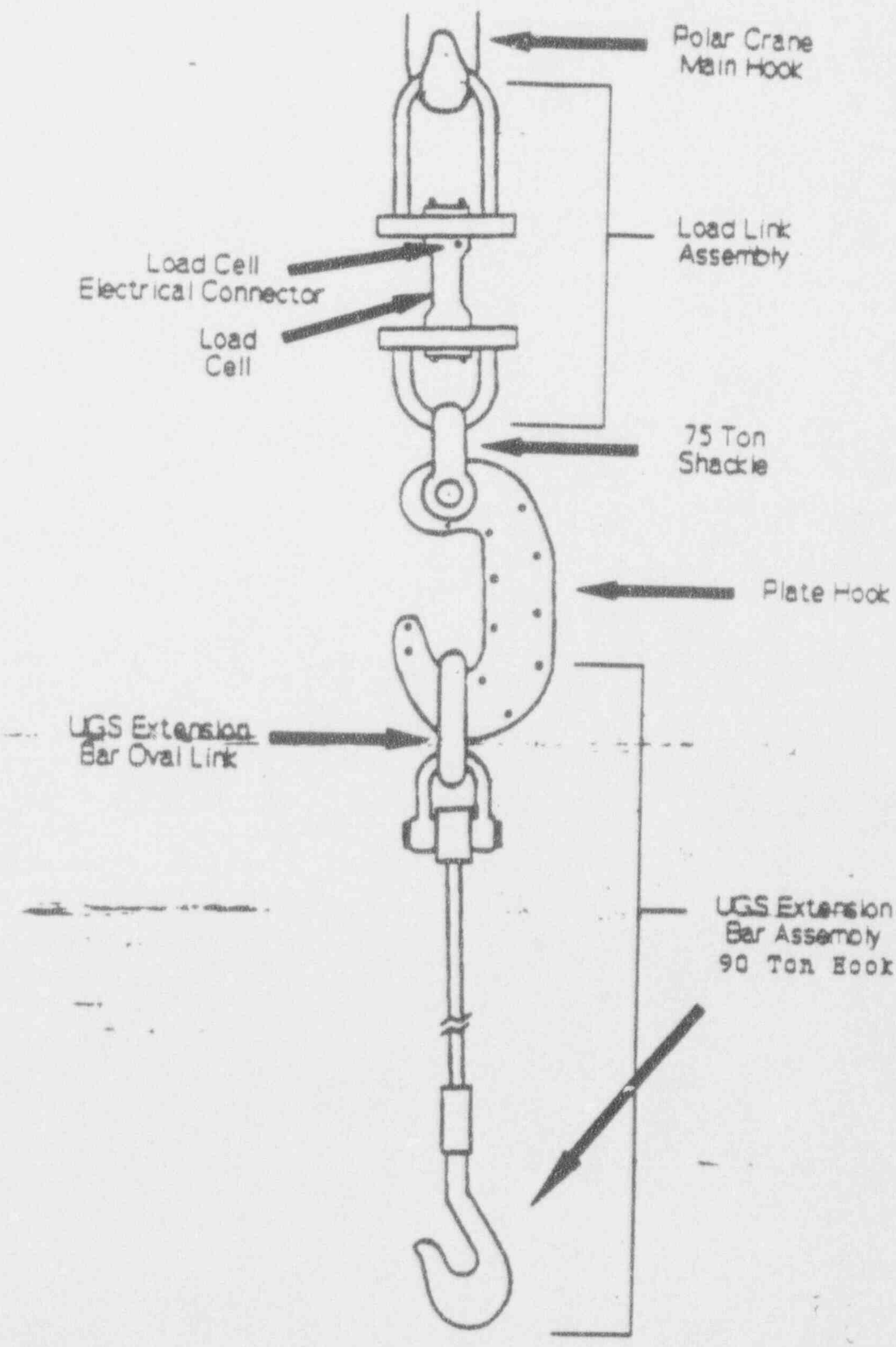
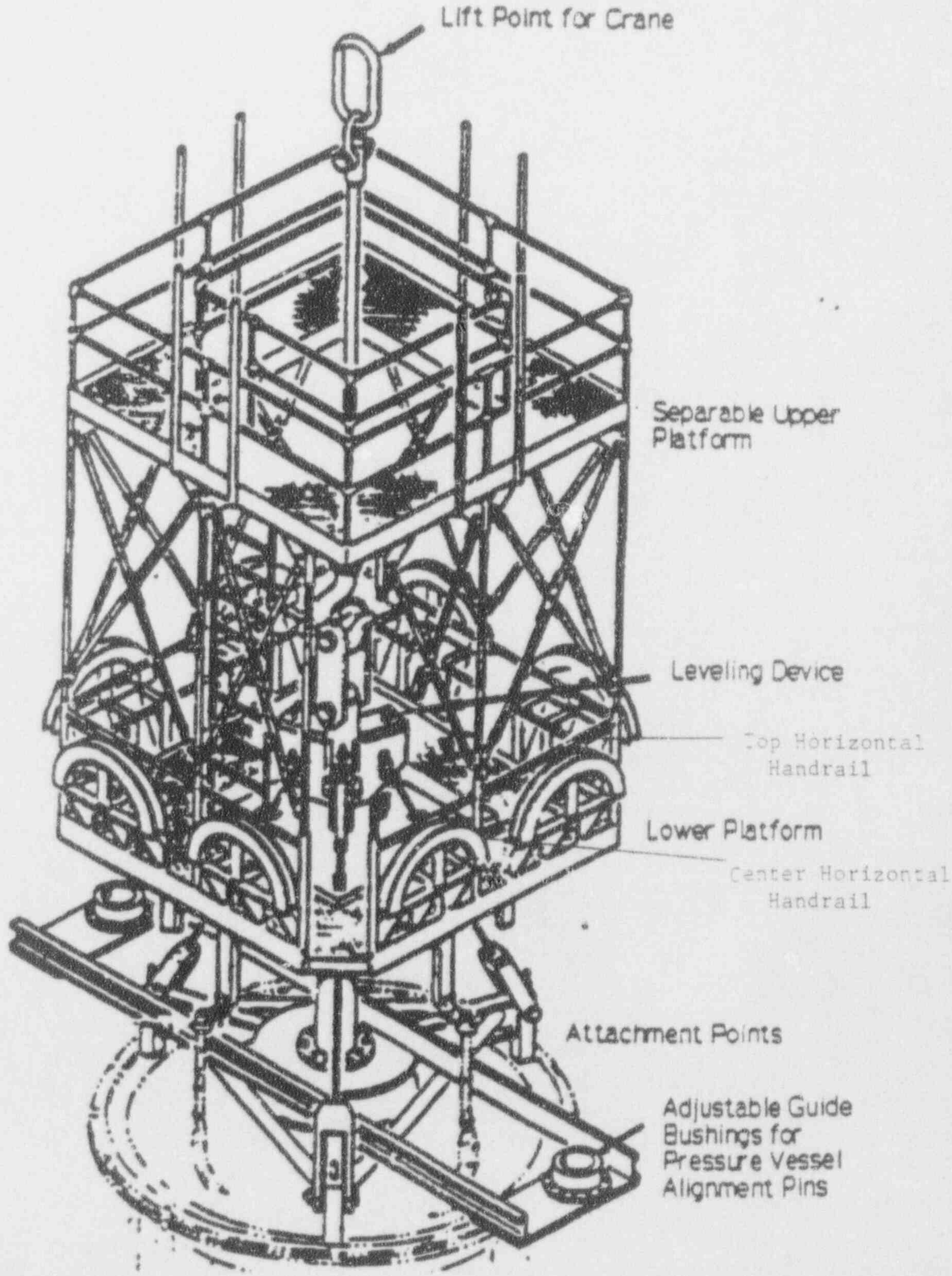


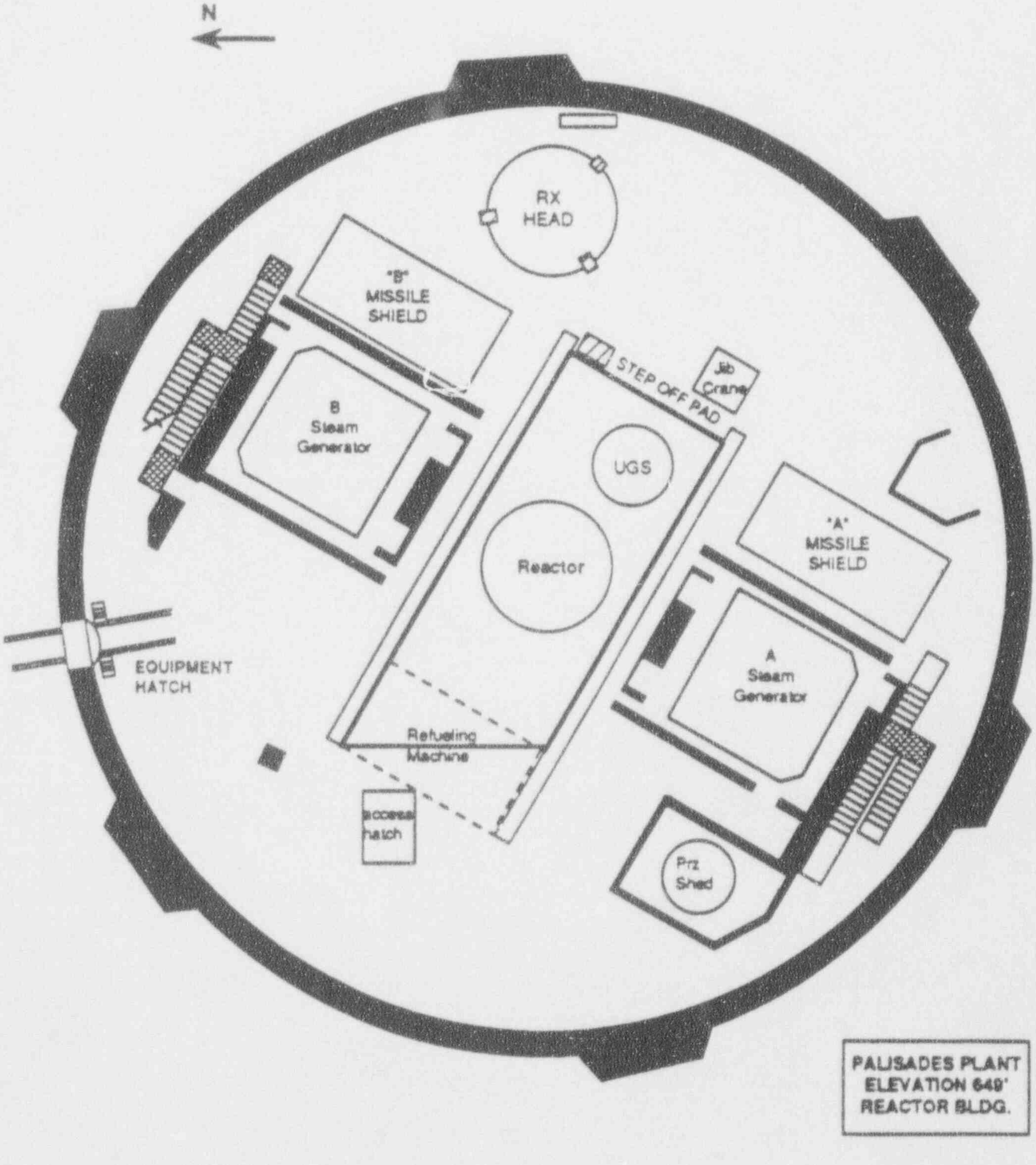
FIGURE 9
UPPER TIE PLATE DIMENSIONS
RELOAD M AND AFTER

LOAD LINKS STACK UP



UPPER GUIDE STRUCTURE LIFT RIG





PALISADES PLANT
ELEVATION 649'
REACTOR BLDG.

September 10, 1993

The Honorable Donald W. Riegle, Jr.
United States Senate
Western Regional Office
Suite 716 Federal Building
110 Michigan Avenue, N. W.
Grand Rapids, MI 49503

Dear Senator Riegle:

This is in response to your July 21, 1993, letter to A. Bert Davis inquiring about a fuel rod breakage identified on July 1, 1993, at Consumers Power Company's Palisades Nuclear Power Plant.

You requested a copy of our investigation report into this matter after it had been completed. An NRC Augmented Inspection Team (AIT) conducted a special inspection at the site on July 8 through 20, 1993. This special onsite review is documented in the attached inspection report.

All our inspection reports are public documents; they are routinely sent to local Public Document Rooms near the subject facilities. In addition, in the case of this AIT inspection, a public meeting was held at the conclusion of the inspection on August 20, 1993. At that time, several members of the public, including your constituents, also requested copies of the AIT report. We are fulfilling each of those requests.

I trust this information is responsive to your needs.

Sincerely,

Original signed by James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosure:
AIT Inspection Report

Distribution:
JTaylor
JSniezek
HThompson
TEMurley
OCA
SECY
Margo (GT 9177)
EDO rf

RIII	EDO	OCA
JMartin	JTaylor	
9/3/93	9/10/93	9/10/93

~~9309160021~~

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

EDO Principal Correspondence Control

FROM: DUE: 08/11/93 EDO CONTROL: 0009177
DOC DT: 07/21/93
FINAL REPLY:

Sen. Donald W. Riegle, Jr.

TO:

Bert Davis, RIII

FOR SIGNATURE OF: ** GRN ** CRC NO:

Executive Director

DESC: ROUTING:

CONSTITUENTS REQUEST RE FUEL ROD BREAKAGE AT THE PALISADES NUCLEAR PLANT

Taylor
Sniezek
Thompson
Blaha
Knubel
Murley, NRR
OCA
SECY

DATE: 07/26/93

ASSIGNED TO: RIII CONTACT: Martin

SPECIAL INSTRUCTIONS OR REMARKS:

REPLY TO GRAND RAPIDS, MI OFFICE.

Cl Lama