U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/93020(DRS)

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

Licensee: Consumers Power Company
Palisades Nuclear Plant
27780 Blue Star Highway
Covert, MI 49043

Facility Name: Palisades Nuclear Plant

Inspection At: Palisades site, Covert, MI

Inspection Conducted: August 19 - 27, and September 29, 1993

Inspectors: Thomas Tongue, Ronald M. Bailey, Michael Parker, Shih Liang Wu, David Nelson, Anthony H. Hsia, Edward D. Kendrick, James L. Javis

Approved By: Bruce L Burgess for
Robert M. Lerch
Team Leader
10/25/93
Date

Bruce Jorgensen
Bruce L. Jorgensen, Chief
Section 2A
Division of Reactor Projects
10/25/93
Date

<u>Inspection Summary</u>

<u>Inspection on August 19 - 27, and September 29, 1993 (Report No. 50-255/93020)</u>

<u>Areas Inspected</u>: Special, announced, team inspection of licensee activities relating to Summer 1993 refueling outage problems, and to plant restart preparations, including: the safety evaluation for the cycle 11 core reload, routine plant operations activities, the impact of the fuel lost from the damaged fuel assembly, the root cause evaluation for the stuck fuel assembly, the regulatory issues identified in the AIT report, and the September 9, 1993 public meeting.

<u>Results</u>: Apparent violations resulting from the AIT inspection are identified in paragraph 7.

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REPORT DETAILS

1.0 Persons Contacted

G. B. Slade, Plant General Manager
T. J. Palmisano, Plant Operations Manager
K. M. Haas, Radiological Services Manager
R. B. Kasper, Maintenance Manager
D. W. Rogers, Safety and Licensing Director
R. M. Rice, Director, Nuclear Performance Assessment Department (NPAD)
K. E. Osborne, System Engineering Manager

These people and others were present at the exit meetings on July 27, and September 29, 1993. R. M. Rice was not present at the September 29, 1993 exit. Other members of the plant staff and NPAD were contacted during the inspection period.

2.0 Introduction

On July 1, 1993, at 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, an Augmented Inspection Team (AIT) was sent 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. A public AIT exit meeting was held with plant management on July 20, 1993. The results of the AIT inspection were documented in Inspection Report 50-255/93018.

After the AIT exit, the NRC continued to monitor the licensee's corrective actions in several ways. Daily briefings on the licensee's root cause analyses and recovery efforts were conducted via conference calls with the NRC until two teams were sent to the site. One team reviewed the safety evaluation for the core reload, which uses reconstituted L-series fuel assemblies - see paragraph 3. A technical review of the of the root cause determinations for the fuel assembly stuck to the UGS was conducted - see paragraph 4. The second team evaluated the plant staff in the performance of routine activities and a trial installation of the upper guide structure - see paragraph 5. A contractor with the second team evaluated the licensee's search for the fuel lost from the damaged fuel assembly and the consequences this fuel might have - see paragraph 6. An assessment of the findings from the AIT report was conducted to determine which ones represented regulatory issues - see paragraph 7. A public meeting was held with the licensee on September 9, 1993 - see paragraph 8.

3.0 The 50.59 Safety Evaluation of the Cycle 11 Core Reload

The NRR team reviewed the licensee's performance in implementing the requirements set forth in 10 CFR 50.59. The principal measure of this performance is the quality of the 50.59 safety evaluation prepared by the licensee for the revised Cycle 11 core design which included the 16 reconstituted L-assemblies in the corner core locations. Also included in the assessment was the documentation of the 50.59 safety evaluation, the licensee's procedure for safety evaluations, and the PRC review process. The review of the 50.59 safety evaluation included its scope, accuracy, and thoroughness in both technical content and documentation.

As a result of the root cause analysis for the damaged fuel assembly, the licensee revised the original Cycle 11 core design by replacing all the I-series fuel assemblies with thrice-burned L- series assemblies reconstituted with 14 stainless steel rods (see Section 3.1 for detailed descriptions). These modified L-series assemblies were located at the 16 corner core locations as shown in Figure 2. According to Palisades Procedure No.3.07, Rev.7, the licensee performed a safety evaluation of the revised Cycle 11 core design in compliance with the regulations set forth in 10 CFR 50.59. Their evaluation was documented in "Palisades Nuclear Plant Safety Review", Plant Safety and Licensing (PS&L) Log No.93-1025, which concluded that there were no unreviewed safety questions in accordance with 10 CFR 50.59. On August 19, 1993, the Plant Review Committee (PRC) reviewed the 50.59 safety evaluation, agreed with the conclusions and approved the revised Cycle 11 core design.

From the perspective of mechanical design, material, core physics, and thermal hydraulics, the inspection focused on the licensee's evaluations of safety significance and whether there was any unreviewed safety question stemming from the revised Cycle 11 core design. The inspectors also assessed the licensee's root cause analysis to date, its plans for continued root cause analysis, and the changes made to address the root causes. The team also agreed with the Palisades PRC conclusion that the revised Cycle 11 core design does not constitute an unreviewed safety question and its approval of the revised cycle 11 core design.

3.1 Description of Changes in the Facility from that Described in the Final Safety Analysis Report (FSAR)

3.1.1 Fuel Assemblies

The M-assemblies, N-assemblies, O-assemblies, and SAN-assemblies have not been altered for the revised Cycle 11. The L-assemblies have been reconstituted for the revised Cycle 11 by replacing 14 fuel rods with 14 solid stainless steel rods. Eight stainless steel rods were placed in the corner of the fuel assembly in the shroud corner with the stainless steel rods placed from the guide bar to the corner plus the location next to the corner location as shown in Figure 1. Each remaining corner of the fuel assembly contains 2 stainless steel rods as shown in Figure 1.

The criteria used by the licensee to select the L-assemblies to replace the I-Hafnium assemblies were:

1. Burnup < 37,500 MWD/MTU at EOC 11 for J-assemblies and K-assemblies.

Licensee Justification: This criteria is somewhat arbitrary and less important if the L- assemblies are used. The objective is to stay below the I-24 assembly burnup at the start of cycle 10 when the I-24 assembly was intact. The L-assemblies have the spacers manufactured with a vertical roll and will grow in that direction with fluence. The spacer cells will not get bigger, rather they will get taller; so, the design limit of 46,000 MWT/MTU is still applicable for the L-assemblies for cycle 11.

2. Assemblies can not be used that have previously occupied one of the eight octant symmetric corner shroud positions.

Licensee Justification: This requirement may be conservative for all eight octant positions. However, it is advantageous to avoid assemblies that previously occupied B-19 and X-19 core locations.

3. Consider assembly bowing to avoid largely bowed assemblies.

Licensee Justification: Assembly to shroud interference already existed at Cycle 10 B-19 and X-19 locations. Placing the bowed assembly towards the shroud in Cycle 11 may worsen the problem.

4. L-assemblies may have some advantages over the J-assemblies and the K-assemblies.

Licensee Justification: The spacer plate material for the L-assemblies was stamped in a preferential direction to minimize cell opening due to growth. Also, using L-assemblies will not require ultrasonic examination since they were examined after their last cycle. The J-assemblies and K-assemblies will require ultrasonic examination prior to use in Cycle 11.

5. Fluence rates values should not exceed Cycle 9 values.

Licensee Justification: Cycle 9 fluence rates were used for consistency with the pressurized thermal shock data currently being reviewed by the NRC.

3.2 Mechanical Design

The stainless steel rods are 0.437 inches in diameter while the fuel rods they replaced were 0.417 inches in diameter. The licensee proposed that the larger diameter rods will reduce the probability of fretting damage during Cycle 11 due to an estimated increase in spring force of 0.6 pounds. The licensee proposed that the corners with 2 stainless steel rods will be less likely to fret and that the load on adjacent fuel rods will be symmetrically increased, decreasing the probability of fretting on the adjacent fuel rods.

The revised L-assemblies and SAN-assemblies have standard design spacers and all of the remaining assemblies in the core for cycle 11 have the debris resistant, High Thermal Performance (HTP) spacers. The I-assemblies that the revised L-assemblies replaced also had the standard design spacers. Hence, the licensee concluded that the use of the revised L-assemblies will not affect spacer induced flow conditions in the core for the revised Cycle 11.

3.3 Core Physics Design

The revised Cycle 11 core contains 52 twice-burned M-assemblies, 68 once-burned N-assemblies, 60 fresh O-assemblies, 8 once-burned N Shield Assemblies (SAN), and 16 modified thrice-burned L-assemblies with 14 solid stainless steel rods inserted in the assembly corners. The difference between the revised and the original Cycle 11 core designs is the 16 reconstituted L-assemblies and the relocation of the selected M and N assemblies (20 assemblies in each batch). Figure 2 shows the revised Cycle 11 core design.

For fluence reduction purposes, the revised Cycle 11 core is a low radial leakage core incorporating 8 SAN assemblies in the flat peripheral regions and 16 reconstituted L-assemblies in the core corner locations. A higher initial reactor coolant boron concentration is needed to offset the additional reactivity in the revised Cycle 11 because of the revised L-assemblies (which are more reactive as compared to the I-assemblies), because the O-assemblies have enrichment higher than previous fuel assemblies, and because of less gadolinia used in Cycle 11 fuels as compared to previous cycles.

3.4 Impact on the FSAR and the Technical Specifications (TS)

The licensee's 50.59 safety evaluation as documented in PS&L Log No. 93-1025 indicates that many FSAR Sections and TS Sections have been reviewed for possible changes due to the revised Cycle 11 core design. These reviews identified FSAR Sections 3.3.1, and 3.3.4.3, and Tables 1-2, and 3-11 to be affected by the design changes. None of the TS Sections was identified to be affected by the design changes.

The licensee plans to update FSAR Sections 3.3.1 and 3.3.4.3 to delete the discussions on the boron carbide neutron absorber rods and the Hafnium clusters that are no longer used, and to add the discussions on the 16 modified L-assemblies and the new assemblies with debris resistant features. The licensee also plans to update Table 1-2 to include Batch O average U-235 enrichment, and to update Table 3-11 for inclusion of L-assemblies and deletion of boron carbide neutron absorber rods and Hafnium clusters.

The staff review also identified FSAR Section 3.3.2.6 which discusses the H-assemblies with stainless steel rods in Cycle 8, the I-assemblies with Hafnium clusters in Cycle 9, and the SAN assemblies in Cycle 10, as used by the licensee to reduce neutron fluence on the reactor vessel. This section should have been identified by the licensee for deletion of the Hafnium clusters and inclusion of the revised L-assemblies. The licensee plans to make these additional changes to the FSAR.

The Licensee's 50.59 safety evaluation stated that the radial peaking factor limits listed in TS Table 3.23-2 were not changed by the revised L-assemblies used to replace the I-assemblies. Although Table 3.23-2 does not specifically list the limits for the revised L-assemblies and the SAN assemblies, Table 6.1 in the Siemens Power Corporation (SPC) report EMF-92-177, Rev.2, implies that the "Revised L" and "SAN" assemblies are bounded by the "M and earlier" and "N" assemblies, respectively. The licensee uses the PIDAL code to perform its weekly monitoring of the radial peaking factors for the 5 different batches of fuel assemblies (Revised L, M, N, SAN, and O).

The NRC evaluation of the FSAR and TS Sections, in light of the revised Cycle 11 design, found it necessary for the licensee to amend the TS Tables 3.23-1 and 3.23-2 to include the different assemblies in the core prior to operation above 25% power.

3.5 Root Cause Analysis Evaluation for I-Assembly Failure

The licensee had proposed that there were 8 potential root causes for the fuel failure in the I-24 assembly discovered after cycle 10. These were:

1. Damaged during fuel moves during previous cycle(s);
2. Damaged during EOC 9 Ultrasonic Test Inspection;
3. Damaged during fuel moves in this refueling outage;
4. Fuel failure due to a loose spacer grid;
5. Fuel failure due to increased primary coolant system (PCS) flow;
6. Fuel failure due to core barrel vibration;
7. Fuel failure due to a manufacturing defect; and,
8. Fuel failure due to a shroud/fuel assembly interface problem.

Number 1 was eliminated because all of the assemblies placed in the core for cycle 10 were ultrasonically tested (UT) between cycle 9 and cycle 10. Number 2 was eliminated since the records indicate that the I-24 bundle was not rotated during UT examination and could not have been damaged by the UT test rig. Number 3 was eliminated by reexamination of the chemistry results for cycle 10. Number 4 was eliminated by the licensee as the root cause but is thought to be a contributing factor. The licensee considers numbers 5, 6, and 7 to be less likely as the root cause. However, the licensee has proposed that 5 and 6 may be contributing factors. The licensee has proposed that number 8 is the most likely cause of the I-24 assembly rod failure.

The use of stainless steel rods in the corners of the replacement assemblies addresses possibility 4, according to the licensee, by increasing the spring force on these rods. The use of stainless steel rods in the replacement assemblies addresses possibilities 5, 6, and 8 by placing sacrificial rods that do not contain fuel in locations where damage occurred to some rods during cycle 10 and which are more susceptible to potential damages in future fuel cycles.

The licensee did not think that the use of stainless steel rods would prevent the type of spacer fretting that occurred on I-24 during cycle 10. However, the licensee proposed that the stainless steel rods would be the only rods affected and no fuel would be lost as a result of spacer fretting.

The team did not complete its assessment of the root cause analysis since the licensee had not finalized the root cause analysis.

Additional data such as examination of core vibrational data, fuel failure monitoring, end of cycle (EOC) 11 fuel assemblies examination, and EOC 11 shroud examination will contribute to the understanding of the root causes. This will be tracked as inspection follow-up item (IFI 50-255/93020-01)

3.6 Technical Evaluations

3.6.1 Mechanical Design

3.6.1.1 Fuel Reconstitution and Generic Letter (GL) 90-02 Supplement 1

The NRC staff issued GL 90-02, Supplement 1, to address the nuclear industry trend of reconstituting fuel assemblies with dummy (nonfueled) rods of stainless steel or zircaloy. The use of dummy rods facilitates the replacement of failed fuel rods when leakers are detected. GL 90-02, Supplement 1 requires that NRC-approved methodologies be applied for reconstitution to ensure compliance with General Design Criteria (GDC) 10.

The staff is reviewing SPC topical report ANF-90-082, entitled "Application of ANF Design Methodology for Fuel Assembly Reconstitution." This report is expected to be approved soon. The licensee's reconstitution for the revised Cycle 11 core is not covered explicitly by ANF-90-082, as discussed in licensee's 50.59 evaluation. However, SPC reanalyzed the Cycle 11 core with the 16 new reconstituted assemblies as described in the safety analysis report EMF-92-177, Revision 2. The results showed that the Chapter 15 analyses were still bounded by the previous analyses. The team considers the reanalyses of Cycle 11 core adequate.

3.6.1.2 Oversized Stainless Steel Rods

The stainless steel rods in the reconstituted L assemblies are slightly larger in diameter than the fuel rods. The larger stainless steel rods in the corner will exert more force on the lantern spring, thereby tightening up the adjacent fuel rods. This results in higher spring force on fuel rods to compensate the spring relaxation during irradiation. The fuel vendor's analysis and testing results confirmed that there was higher spring force on the adjacent fuel rods. Since SPC has confirmed that the bimetallic spring relaxed as expected, the use of larger stainless steel rods tends to reduce the potential for fretting between the fuel rods and spacer springs. Thus, the team considered the use of larger stainless steel rods acceptable.

3.6.1.3 Spring Retention

The rod withdrawal force data from fuel assemblies I, J, K, H, and L have shown that the spring force relaxed significantly during Cycle 10. In some cases, the rod cell force was completely diminished. However, this result was not unexpected by SPC.

SPC incorporated the Palisades data into the spring force relaxation versus burnup data from other SPC assemblies. The results showed that the bimetallic spring relaxed with higher burnup as expected for Palisades' spacer springs. SPC's analysis has demonstrated that the spring force will remain higher than the vibration force until the end of Cycle 11. The team understands that the licensee's current fuel design uses the high thermal performance (HTP) grid spacer. The HTP grid spacers do not use bimetallic springs and should be less prone to the fretting damage. Because the Palisades spring data were within the bounds of the analysis and the larger stainless steel rods were used to increase spring retention, the team concludes that the spring retention has been adequately addressed.

3.6.1.4 Fretting Wear Against Core Shroud

Although the licensee has not yet determined the root cause, some possible causes were surveyed and examined. The licensee speculated that the most probable cause was due to interference between the core shroud and fuel assembly, indicated by the missing or torn grid spacers and the corresponding wear indications on the core shroud. While the Palisades fuel failure root cause analysis is continuing, the licensee has taken actions to prevent such a phenomenon from recurring by placing stainless steel corner rods in the reconstituted L-assemblies which will be placed in the core corner locations. This is a solution that is similar to that used in resolving the Westinghouse reactor baffle jetting problem. The reconstituted L-assemblies will reside in the Cycle 11 core for only one cycle, and will be replaced by new shielding assemblies in future reloads. Based on the reconstituted assemblies, the team concludes that the licensee has taken appropriate action in mitigating the consequence of interference between fuel assembly and core shroud.

3.6.1.5 Wear of the Control Rods

The revised L-assemblies are placed in the core with fluence induced bow toward the center of the core. The licensee has proposed that orienting the revised L-assemblies in this manner minimizes the possibility of interactions of the fuel assemblies with the shroud corner. The revised L-assemblies were selected based on those L-assemblies with the most uniform fluence. The maximum projected burnup gradient across the revised L-assembly at the end of Cycle 11 was calculated by the licensee to be approximately 6,400 MWD/MTU. Eight of the revised L-assemblies are next to a control rod. The maximum projected gradient for an L-assembly next to a control rod is calculated to be about 4,600 MWD/MTU. The licensee reports that during previous cycles, assemblies with gradients as high as 7,000 MWD/MTU were oriented towards a control rod resulting in no interaction between the assembly and the control rod. The licensee proposes that the burnup gradients for the revised L-assemblies is expected to be bounded by past operating history of SPC fuel at Palisades.

The space between fuel assemblies is 0.365 inches and the width of the blades on the control rod is 0.180 inches leaving a gap of 0.093 inches between the fuel assembly and the control rod blade on each side of the blade. The licensee has proposed that the bow in the assembly will be less that 0.093 inches based on past operating history and calculations by the licensee's fuel supplier. The NRC team concurs with the licensee's analysis that the fuel assemblies next to the control rods will not affect the functionality of the control rods.

3.6.2 Core Physics Design

As stated in the SPC Letter, to the licensee dated August 17, 1993, HGS:312:93 the evaluation of the core physics design changes is documented in Revision 2 (August 17, 1993) of EMF-92-177, "Palisades Cycle 11 Safety Analysis Report," and in Revision 1 (August 23, 1993) of EMF-92-178, "Palisades Cycle 11: Disposition and Analysis of Standard Review Plan Chapter 15 Events,"

The physics characteristics evaluated in Revision 2 of EMF-92-177 include power distribution, control rod reactivity, and moderator temperature coefficient (MTC) considerations. The Cycle 11 core loading configuration was redesigned by SPC to minimize radial peaking factors; specifically, to remain within the approved Technical Specification (TS) limits of:

- Assembly Radial Peaking Factor, F_r^A limit of 1.76 (Batch O), 1.66 (Batch N), and 1.57 (Batches M and L)
- Total Radial Peaking Factor, F_r^T limit of 2.04 (Batch O), and 1.92 (all others)
- Linear Heat Generation Rate (LHGR) limit of 15.28 kw/ft to 60% core height; linearly decreasing to 14.21 kw/ft at 100% core height

The maximum calculated values for F_r^A and F_r^T were 1.575 and 1.851, respectively; which, when combined with the TS uncertainties, are within the TS power peaking limits. The largest calculated LHGR is 11.63 kw/ft which is also within the TS limiting value with the TS uncertainty included.

Shutdown margin calculations were performed for the modified Cycle 11 configuration, yielding a cycle minimum shutdown margin of 2.56% delta-rho at End of Cycle (EOC), Hot Full Power (HFP) conditions, which is above the TS low limit of 2.00%. The moderator temperature coefficient (MTC) was evaluated for the revised Cycle 11 core configuration at both Hot Zero Power (HZP) and HFP for Beginning of Cycle (BOC) and EOC conditions. The calculated MTC values are within the safety analysis limits of $+0.5\times10^{-4}$ delta-rho/degree F and -3.5×10^{-4} delta-rho/degree F.

The minimum departure from nucleate boiling ratio (MDNBR) evaluation for the revised Cycle 11 core configuration was performed with the ANFP CHF correlation for the TS limiting radial peaking factor values. The re-analysis of eight Standard Review Plan (SRP) Chapter 15 events, as reported in Revision 1 of EMF-92-178, shows that calculated DNB margins were improved with the exception of Events 15.4.2 - Uncontrolled control rod bank withdrawal at power, 15.4.3(5) - Control rod misoperation: single rod withdrawal, and 15.6.1 - Inadvertent opening of pressure relief valve, which showed a slight DNB margin degradation. All FSAR event acceptance criteria, however, were met for the revised Cycle 11 core.

3.6.3. Assembly Flow Vibration And Core Barrel Vibration

The licensee considered core barrel vibration and assembly flow vibration as two possible contributing factors to the fretting wear problem. The core barrel vibration problem was discovered early in the plant life and was addressed by adding a hold-down ring to the UGS. There was no indication in subsequent operations that the core barrel vibration occurred again. The licensee included the core barrel vibration in its root cause analysis and concluded that it was a possible contributing factor. The team considered the licensee's effort in addressing the concern of core barrel vibration adequate.

For the assembly flow vibration, the licensee has reconstituted the L-assemblies with larger stainless steel rods in the corner locations. The reconstituted assemblies are designed to mitigate the potential fuel rod damage due to assembly flow vibrations. The team considered the licensee's effort in addressing the concern of assembly flow vibration adequate.

Recently, a few other PWR plants also experienced similar phenomena of fuel failure and fretting wear near the core shroud. The assemblies involved were of a newer spacer design located near the core shroud. Subsequent examinations and flow testing uncovered that there was a natural vibrational frequency that existed for a combined condition of a particular spacer design and restricted flow. Those licensees involved have modified the fuel assemblies to dampen the flow vibration for the short term. The fuel design and flow testing procedures have been modified to take into account this effect. Since this phenomenon is rather new and unique, the staff is preparing an Information Notice to alert all licensees of this type of flow induced vibrational fretting.

The effects of the modified L-assembly oversized stainless steel rods on local assembly flow distributions and the intra-assembly and inter-assembly cross flows was evaluated by SPC and was found to be insignificant. The effects of the revised core loading configuration on the core-wide flow distribution were analyzed by SPC using the XTG computer code, including the simulation of the two spacer designs (bi-metallic for the L-assemblies and HTP for the remaining assemblies). The results showed that the revised core design had insignificant effects on the core wide flow distribution.

3.6.4 Materials

The licensee analyzed the irradiation-induced spring force relaxation. The spacer springs are designed such that they will not damage the cladding during installation nor will they damage the cladding during operation due to differential thermal expansion. The spring force should be sufficiently high to prevent fretting of the cladding, and to suppress as-fabricated and thermal bow of the fuel rods and to resist flow induced vibration of the fuel rods.

The springs start with a spring force ranging from 2.6 to 4.5 pounds. The minimum spring force is the force required to overcome flow induced vibration force, which is 0.08 pounds. The minimum spring force at the end of life will be a 90 percent relaxation of the 2.35 pound beginning of life spring force or 0.2 pounds, which exceeds the flow induced vibration force of 0.08 pounds.

The team reviewed the licensee's submittal dated August 16, 1993, responding to the NRC request for additional information on the fuel failure event, and had the following comments. Examination of the I-assembly fuel rod withdrawal data showed that many of the individual spacers had apparent spring forces less than 0.20 pounds with many of the spring forces being 0.0 pounds. There was no evidence of fretting damage at many of the spacers with low recorded spring forces. On the other hand, there were spacers with spring forces of 1.43, 1.32, 0.69, and 0.59 pounds that gave minor eddy current indications, and spacers with spring forces of 0.69, 0.68, 0.59, 0.48, 0.34, and 0.29 pounds that gave severe eddy current indications. Eddy current indications are indicative of fretting damage. The average spring forces for all of the spacers appear to confirm the licensee's analysis; however, individual spacer data are contradictory.

The average Palisades rod withdrawal force relaxation versus burnup data agrees with data from other SPC assemblies. The team agreed with the licensee's determination of a 90 percent relaxation of spring forces at end of life as being conservative based on the data presented.

The data presented by the licensee indicates that fast flux exposure for a fourth cycle on the reconstituted L-assemblies will not affect the performance of the assemblies. The licensee does not intend to use the reconstituted L-assemblies past cycle 11. The team concurred that fast flux exposure for a fourth cycle will not adversely affect the L-assemblies. Furthermore, the preferential stamping of the L-assemblies grid spacers so that the cells will not increase in size with exposure is a marked improvement for the reconstituted L-assemblies over earlier assembly designs.

3.6.5 Burnup Considerations

Initially, the I-shielding assemblies were planned for three cycles (Cycles 9, 10, and 11) in the core periphery. The I-assemblies would have been in the core for a total of six cycles, taking into account the previous three cycles of normal operation. During the Cycle 9 outage, the licensee examined the whole core with ultrasonic testing (UT). There were no leakers among the I-assemblies. During Cycle 10 operation, some I-assemblies were damaged by the flow vibrational fretting. Thus, it is prudent practice that, after three cycles of normal power operation, an assembly should not go beyond one more cycle (the fourth cycle) when used for shielding. The licensee plans to use the reconstituted L-assemblies for only one cycle, Cycle 11, and plans to replace the reconstituted L-assemblies with a new design of shielding assemblies for future reloads.

The selection criteria used for replacement candidates for the I-assemblies included a target EOC 11 burnup for J and K assemblies of less than the BOC 10 burnup for assembly I-024 (37,500 MWD/MTU). The assemblies finally chosen were from the L-assemblies with an allowable maximum burnup limit of 46,000 MWD/MTU due to the improved grid spacer design. The projected batch average EOC 11 exposure for the L-assemblies is approximately 36,500 MWD/MTU (with a range of 33,321 to 40,028). Thus, the team concluded that, based on burnup considerations, the reconstituted L-assemblies were acceptable.

3.6.6 Fluence Reduction

The design criteria for the revised Cycle 11 core loading configuration includes a requirement that fluence rates not exceed the documented Cycle 9 values. According to the Palisades reactor engineering staff, the criteria used by SPC to meet this requirement was that the bundle powers in the peripheral core locations be less than or equal to the Cycle 9 core peripheral bundle powers. A preliminary scoping analysis was then performed by Palisades staff using their in-house DOT-IV model which confirmed that the corresponding fast fluence (>1.0 Mev) values were less than the Cycle 9 values at the critical weld orientations. The final fluence evaluation calculations will be performed by Westinghouse, to be consistent with the Cycle 9 results previously furnished to the NRC.

3.7 Plant Review Committee (PRC) Review and Approval

The PRC reviewed and approved the 10 CFR 50.59 submittal on the modified core reload plan on August 19, 1993. The PRC review was observed by members of the NRC team. The Palisades team that prepared the 10 CFR 50.59 review were questioned extensively by the PRC personnel prior to receiving approval. Questions were raised about the chromium plating on the SAN-8 upper tie plate alignment pin hole inside diameter, about the cycle 11 fuel load integrity, about the plans to detect a low power fuel rod failure during cycle 11, about the plans to determine if the shroud is interacting with fuel assemblies during cycle 11, and numerous additional questions. All of the questions were addressed adequately by the licensee staff.

3.8 Documentation of the Licensee's 50.59 Safety Evaluation

The 50.59 safety evaluation was documented according to Procedure No.3.07, Rev. 7, "Safety Evaluations". However, the audit team found the 50.59 evaluation package designation confusing in that the current revised Cycle 11 core was identified as FC-934, Rev. 0, with an associated safety evaluation SE Rev. 1 whereas the original Cycle 11 core was also identified as FC-934, Rev. 0, with an associated safety evaluation SE Rev. 0. The licensee indicated that the difference between these Cycle 11 designs was documented in an Engineering Design Change (EDC). The existence of the associated EDC was not mentioned in the current FC-934, Rev. 0 documentation for the reader to realize that the current FC-934, Rev.0 is different from the original Cycle 11 core design bearing the same Item Identification Number.

The team reviewed the documentation of the 50.59 safety evaluation for its technical depth and thoroughness and found it acceptable but improvements are needed. The licensee's 50.59 evaluation adequately documented the revised Cycle 11 core physics design, the various assembly mechanical designs, I-assembly replacement criteria, detailed L-assembly modifications, assembly reconstitution considerations, qualitative thermal hydraulic DNB considerations, and the evaluation of the projected bowing from the revised L-assemblies.

The documentation should have included or referenced the thermal hydraulic analysis addressing the core average and subchannel flows as a result of the new design, its impact on core barrel and assembly vibrations; the loose grid spacer, spacer spring retention forces and its impact on fretting wear; and a discussion of how the fluence reduction criteria are satisfied by the new design. These issues were addressed in other documentation as discussed above.

3.9 Conclusions

The team completed its audit review of the licensee's 50.59 safety evaluation for the revised Cycle 11 core design and concluded that the licensee followed their procedure in performing the safety evaluation and the evaluation and its associated documentation are acceptable. The team also agreed with the Palisades PRC conclusion that the revised Cycle 11 core design does not constitute an unreviewed safety question and its approval of the revised cycle 11 core design.

No violations, deviations, unresolved or inspector followup items were identified.

4.0 Stuck Fuel Assembly Root Cause Evaluation

On July 6, 1993, fuel assembly SAN-8 was inadvertently partially lifted with the upper guide structure (UGS) from core position Z-11. Two previous incidents involving inadvertent lifting of a fuel assembly from core position Z-11 with the UGS were experienced in 1988 and 1992. Initial followup of the July 6, 1993, stuck fuel assembly was performed by the resident inspectors and documented in inspection report No 50-255/93017(DRP). Subsequent followup of the event was performed by an augmented inspection team (AIT) and documented in inspection report No. 50-255/93018. The AIT performed an extensive review of the event including evaluation of the root cause for the stuck fuel assembly. At the conclusion of the AIT inspection the licensee had not identified any single root cause for the stuck assembly; however, several potential contributors had been identified.

The potential contributors to the lifting of the stuck fuel assembly with the UGS were:

- Undersized upper tie plate pin holes in SAN-8.

- Deformation of core shroud creating an interference between the UGS fuel alignment pins and the fuel assembly.

- Fuel assembly bow.

- UGS fuel alignment pins loose or returned to a bent state during operation.

- UGS fuel alignment pins out of position or the core support plate alignment holes out of position.

- Debris causing hang-up of the fuel assembly in the UGS.

- UGS not level during lift, to the extent that there was interference between the UGS and the fuel assembly.

- Core support barrel mis-located.

- Damage to alignment pins or to the lower alignment plate lifting or setting of UGS.

- Degraded surface condition of the UGS alignment pins at core location Z-11 which could have promoted sticking within fuel assembly upper tie plate holes.

- Loss of preload on cap screws and alignment pins, which hold the UGS together, resulting in a significant loss of structural rigidity.

The licensee developed an action plan to determine the role of each of the potential contributors, during the lifting of the fuel assembly with the UGS. As a result, the licensee identified 75 action items relating to the 11 potential contributors, for further followup.

Followup inspection on the action items has resulted in the licensee's root cause investigation team identifying the following:

- UGS fuel alignment plate pins bent in location Z-11N (1.56 degrees), Z-11S (0.41 degrees), and Z-16S (0.99 degrees).

- A single gage of 0.995" diameter could not be placed on pins Z-11N and Z-16S, indicating the pins to be curved.

- Analyses indicated that angular misalignment between UGS fuel alignment pins and fuel assembly upper tie plate of 1-1.5 degrees was sufficient to lift fuel assemblies.

- Qualification test results and analyses indicate that bent and straightened pins have a gap between alignment pin shoulder and the UGS lower alignment plate. Bent and straightened pins are less rigid and less resistant to bending than originally installed pins.

- SAN-08 upper tie plate had two distinctive peen marks offset from the center of upper tie plate holes by approximately 0.5 inch. Peen marks are in the orientation and separated by about the spacing of UGS fuel alignment plate pins.

- SAN-08 video camera inspection in the spent fuel pool identified a piece of debris stuck to the bottom of one foot. Debris could have made it harder to fully seat SAN-08. An object 0.05 inch thick under an assembly foot would dislocate the upper tie plate by approximately 0.5".

- With only SAN-08 removed from location Z-11, the core support plate was inspected with no debris identified. Several days later, with five assemblies removed from around Z-11 location, debris was observed.

- Reactor head alignment pin at 0 degree location was incorrectly installed.

- Measurement of UGS levelness on successive lifts indicates significant variation of direction and magnitude of out-of-levelness.

- Analysis indicated assembly bow may contribute to angular misalignment between UGS alignment pins and upper tie plate.

- Close up video camera inspection of UGS alignment pins at location Z-11 indicated a greater degree of engagement in the upper tie plate alignment holes.

- Tip of UGS fuel alignment plate pin at location Z-16S was observed to be distorted during video camera inspection.

The above findings resulted in the following potential contributors to a stuck fuel assembly being eliminated:

- UGS fuel alignment pins out of position or the core support plate alignment holes out of position.

- UGS fuel alignment pins loose or returned to a bent state during operation.

- Deformation of core shroud creating an interference between the UGS fuel alignment pins and the fuel assembly.

- Undersized upper tie plate holes in SAN-8.

- Loss of structural rigidity of the UGS.

- Damage to the UGS fuel alignment plate.

- Debris between the fuel assembly upper tie plate holes and the UGS fuel alignment plate pins.

- Mis-location of core support barrel.

The remaining potential contributors have been categorized as to their significance in contributing to the stuck bundle.

Potential contributors found highly likely to solely result in bent fuel alignment pins and/or stuck fuel assemblies were:

- Tilted/unlevel UGS.

- UGS with bent fuel alignment pins.

- Fuel assembly not properly positioned or seated.

Potential contributors having moderate magnitudes and probabilities and which could combine with other contributors to bend fuel alignment pins and/or stick fuel assemblies were:

- Fuel assembly bow.

- Fuel alignment pins with degraded surface conditions.

The licensee has taken the following actions to specifically address the five remaining potential contributors identified above:

- Replaced UGS fuel alignment plate pins Z-11N, Z-11S, and Z-16S.

- Replaced the fuel assembly (SAN-08) upper tie plate for core location Z-11. The upper tie plate modified design reduces the potential for interference between tie plate alignment holes and UGS alignment pins.

- Implemented methods to evaluate fuel assembly elevations and relative positions after reactor core reloads.

- Modified equipment and revised procedures to assure UGS lift rig/UGS levelness is established.

- Modified procedures to assure UGS is level within acceptable limits prior to lift of UGS.

In reviewing the licensee's root cause analysis, corrective actions to prevent reoccurrence, and trial insertion/removal, the inspectors noted that the licensee had expended significant resources to address the root cause for the fuel assembly lifting. The licensee's root cause investigation team was composed of highly dedicated individuals who utilized extensive problem solving techniques. Management was actively involved throughout the investigation. Although the licensee had not completed all the long term actions, the actions taken were demonstrated adequate for cycle 11 operation. The following long term actions were being evaluated by the licensee.

- UGS levelness.
- Replacement of RV head/UGS alignment pins.
- Minimized fuel assembly bow.
- Centering of crane.
- Fuel assembly height/levelness/core verification.
- Modification of upper tie plate for all fuel assemblies.
- Camera/lightning/water clarity.
- Dedicating specific load cell for UGS removal/insertion.
- Future gauging of fuel alignment pins.
- UGS key/keyway measurements.
- Procedure upgrades, as necessary.

No violations, deviations, unresolved, or inspection followup items were identified in this area.

5.0 Observations of Activities

The team observed selected activities and interviewed licensee personnel to evaluate the effectiveness of communications in the plant staff and workers understanding of management expectations. Overall, the results were positive with regards to employee readiness for operations. At the time of the inspection however, there was limited activity in the plant; the outage work was mostly completed and the licensee was concentrating on finalizing the efforts of the root cause assessment teams and final engineering restart issues.

5.1 Operations

The team observed the conduct of operations personnel both inside and outside of the main control room during major evolutions. This observation included equipment testing, surveillances and maintenance activities being conducted in support of the outage. Based upon these observations, the team concluded that the operations personnel demonstrated a very good awareness of plant conditions and were able to communicate their knowledge through proper coordination and control of plant activities in a safe manner. Additionally, the operation's shift turnover activities were performed in a professional and competent manner which ensured the appropriate transfer of information to oncoming shift personnel.

Due to the low frequency of assigned work orders and surveillance tests in progress, the team was unable to properly evaluate the adverse effects of maintenance and testing activities on the operators' ability to control support activities and maintain safe plant operations. However, a limited number of activities that were observed appeared to be conducted in a controlled manner which did not affect the operators' ability to perform normal duties.

In general, procedures and administrative controls were in place to adequately control and direct the safe startup and continued operation of the plant. The team reviewed selected operations procedures and controls (i.e. tagouts) to verify the current plant conditions with existing procedural requirements with minimal discrepancies noted. This verification included the system walkdown of one safety system tagout and daily plant tours observing equipment status.

The team determined through interviews and observation of on-the-job performance that operations personnel were knowledgeable and capable of performing their licensed duties. Shift manning was maintained in accordance with Technical Specifications at all times.

The team concluded that management actions to improve performance were in place and efforts undertaken to date were effective. The team interviewed selected personnel in the management and non-management staff positions for licensed operators. In general, the non-management position operators felt that management goals, directives and policies were adequately represented to them but the importance of incorporating operator feedback into the improvements was not clear. Management position operators felt that management goals, directives and policies emphasized the professionalism with which they conducted their jobs and were making improvements in plant performance.

5.2 Maintenance

The team observed work in all the maintenance disciplines and saw that work orders and procedures were available, adequate and followed, spare parts and tools were proper and available, and that the knowledge and training of the personnel involved was adequate for the job. Observations were made of shift briefs, and pre-job briefs, which were thorough, with good discussion. Management expectations were made known. Post-maintenance testing was also observed and found to be appropriate. The test results met the acceptance criteria and were properly documented and trended. The team noted the presence and involvement of first line supervisors and system engineers. Interdepartment cooperation was good. NPAD assessors were also present occasionally. In summary, in light of the limited activities in progress, the maintenance observed was performed in an acceptable manner with adequate resources and oversight, and with appropriate consideration for safety.

5.3 Engineering

The team discussed engineering efforts in progress and readiness for startup with the engineering staff. For the issues discussed, the engineering deliberations were thorough and conservative with regard to achieving complete and final resolutions. Nuclear engineers were prepared for startup having identified personnel assignments, training, and the procedural approach to startup testing. It was noted however, at the time of the inspection, that engineering was involved in resolving several significant operability issues the licensee had identified. The issues were being appropriately identified, discussed, and evaluated for corrective action.

5.4 Radiological Services

During the inspection, Radiological Services Department (RSD) performance during the refueling outage was assessed and RSD plans for supporting start-up activities evaluated. In addition, the RSD plan for responding to the presence of fuel in the primary coolant and corresponding systems, once start-up activities begin, was reviewed. Conclusions drawn were based on interviews with RSD management and involved personnel, observations of work activities in radiologically controlled areas and reviews of pertinent documents.

The RSD participation in the outage had four specific elements, each of which was assessed during the inspection.

- RSD planning and scheduling activities both before and during the outage were excellent. Jobs were performed on schedule and the RSD had ample staff to provide coverage when needed. Even after the master schedule was changed, following the stuck control rod incident and the discovery of a failed fuel pin, the RSD planners were able to plan new work requests in a timely manner and work with the schedulers to insure that critical jobs were not delayed.

- RSD pre-job briefs needed improvement. Interviews with individuals who had attended pre-job briefs indicated that there were numerous distractions (doors opening and closing, people talking and phones ringing) in the areas were the briefs were held and the quality of the presentation was dependent on the technician giving the brief. Prior to the inspection, the RSD had been made aware of the deficiencies and had taken steps to correct them. The RSD tried to find quieter areas to hold the briefs and had developed a check-off sheet to be used by the presenter to insure that all the relevant information was passed on to the worker during the brief.

- Radiation safety technician performance during the outage was generally very good. The technicians appeared to be technically competent and well trained. There were, however, problems with contractor technicians setting poor examples for other workers. A Nuclear Performance Assessment Department (NPAD) surveillance reported that a number of contractor technicians had used poor radiological practices. Those practices included improper placement of dosimetry and wearing scrubs in areas where scrubs were not allowed. The poor practices were brought to the attention of RSD management and immediate corrective action was taken. The surveillance concluded that, in general, technicians did a good job of keeping workers informed of radiological conditions, providing advice about radiological conditions, and controlling access to various radiological areas. The inspectors who had worked with the RSD technicians during the inspection concurred with this conclusion.

- Post-job briefs were held in accordance with station procedure and appeared to be effective.

In conclusion, the RSD performance during the outage was effective. The one deficiency identified during the outage, pre-job briefs, had been noted by the RSD and corrective action had been taken.

Following the discovery of a broken fuel pin in the Reactor Cavity Tilt Pit, the RSD developed a RSD Fuel Failure Response Plan to identify, track the status of, and document RSD actions taken as result of the fuel failure. The plan was a living document and once completed would document the basis for the program with regard to failed fuel. The plan contained assumptions made about the risks associated with failed fuel and detailed the organizational structure for implementing the plan. Actions within the plan were assigned to specific individuals and target dates were set for completing the assigned tasks. Of the more than 56 action items identified in the plan a number were directly related to RSD start-up activities and RSD plans for tracking data points to indicate failed fuel in the future. Those actions directly related to start-up activities included:

- Review and revision of the whole body counter's (Fast Scan) library to accommodate fuel material

- Reevaluation of the frequency of surveys during start-up and normal operations

- Reviewed derived air concentration (DAC) calculation and skin dose methodology to account for new fuel nuclide mix

- Evaluation of TLD (Panasonic) algorithms for response to the presence of fuel

- Evaluation of PCM-1b (Personnel Contamination Monitor), PM-7 (Portal Monitor) and frisker energy distribution of calibration sources

- Training the RSD staff in preparation for start-up activities

The whole body counter, PCM-1b and PM-7 and DAC evaluations had been performed and completed. The RSD decided to increase the frequency of surveys during start-up activities, electronic dosimeters would be posted in critical areas throughout the plant and the data collected every four hours during start-up. Following start-up the electronic dosimeters would stay in place and the data they supply tracked.

RSD plans for tracking data points for an indication of failed fuel included:

- Part 61 samples were collected and sent to the licensee's vendor laboratory for analysis. Future analyses would be tracked for the presence of fuel.

- The hot spot program was reevaluated to include the possibility of finding fuel fragments. Criteria for determining when to perform gamma spectral analyses on newly discovered hot spots would be developed. That evaluation was due to be completed by October 31, 1993.

- The licensee would review the data collected from personnel contamination incidents, whole body counts, air samples, and massilin survey smears to determine if it could be used in the future to indicate fuel failure. That evaluated is due to be completed by October 31, 1993.

In conclusion, the response plan was comprehensive in scope and provided a good basis for integrating the RSD response to operational events during start-up activities and implementing plans for tracking various parameters during normal operations for the presence of failed fuel. The RSD appeared fully prepared to support the facility during start-up activities.

The Nuclear Performance Assessment Department performance during the outage with regard to the RSD activities was also assessed. One surveillance and a number of Field Monitor Reports were reviewed. Following field assessment activities, NPAD assessors issue Field Monitor Reports to report their findings. If deficiencies are noted during the assessment they are recorded in the department's database and the NPAD director meets with the plant manager once a week to discuss the previous week's findings. If a deficiency warrants management attention the assessor will normally issue a Deficiency Report. A review of the weekly reports indicated that while many of the deficiencies identified during the assessments had been brought to management's attention, none of them had been documented in the RSD deficiency reporting system (radiological deficiency reports).

NPAD uses its database to record deficiencies and track corrective actions. In principle the RSD is responsible for correcting its own deficiencies, however, if those deficiencies are not reported in the RSD system NPAD assumes that responsibility. For example, during the outage NPAD conducted a surveillance to assess health physics technician performance during backshift. In general, the assessors found that the technicians had demonstrated good performance, however, several technicians were observed using poor radiological practices. In the surveillance, NPAD reported that the RSD had been informed of the observed practices and had taken corrective action. The deficiencies, however, had not been documented in the RSD deficiency system and plant management had not been given a copy of the surveillance. Under this system NPAD, not the RSD, had been responsible for insuring that the deficiencies had been corrected and the actions taken documented. This was a weakness in the program.

In summary, the RSD performance during the outage was good to excellent. The pre-job briefs needed improvement and steps were taken to require use of the briefing check list. The RSD had developed a comprehensive plan in preparation for start-up activities and the plan provided a good basis for supporting those activities. The NPAD system for reporting deficiencies and documenting corrective actions needed improvement.

5.5 Trial installation of the UGS

The licensee decided to perform a trial insertion and removal of the UGS to demonstrate successful removal of the UGS without a fuel assembly. This action was also used to demonstrate/identify the interaction of the UGS with the fuel assemblies and the core support barrel by closely monitoring this activity with underwater cameras. The licensee performed the trial insertion and removal on August 21 and August 22, 1993, respectively. The inspectors observed both the insertion and removal activities. The observations are addressed below.

The inspectors observed the upper guide structure (UGS) set onto the core on August 21, 1993, and also the lift from the core on August 22, 1993. Both activities were performed in accordance with the applicable sections of procedure RVI-M-12, "Final 1993 Installation of Upper Guide Structure."

The inspector attended the pre-job brief for both activities. The briefs were comprehensive, and covered the procedure steps that were to be performed in detail. The inspector verified that all personnel with assigned responsibilities were present.

Proper radiological protective measures were established for personnel once inside containment. Proper dosimetry and protective clothing were worn by each individual, and coverage by the radiation protection technician assigned to monitor the job was good.

The inspector observed that proper communications were established between the control room and the senior reactor operator/shift supervisor directing the evolutions from containment. Cameras, lights, video equipment, and other measuring devices were properly aligned and staged.

The actual lift evolution was satisfactory. The UGS was lifted to the six inch elevation above the top of the fuel assemblies and held for data gathering and visual inspection. During the lift, the fuel alignment plate on the bottom of the UGS was monitored for attached fuel assemblies and none were observed. Particular attention was paid to core location Z-11. The lift of the UGS was smooth and no swaying or tilting of the UGS was observed. Levelness of the UGS was checked acceptable. The load cell indicated that the UGS was within its expected weight.

The UGS was then lifted to the three foot elevation and a thorough camera inspection was performed to verify that there were no attached fuel assemblies. Following this inspection the UGS was transferred to its storage location and set on its pads with no apparent problems.

No violations, deviations, unresolved or inspector followup-items were identified.

6.0 Evaluation of the Fuel Lost from the I-24 Assembly

The Palisades Nuclear Power Plant experienced a fuel failure in assembly I-24 at core location B19 during cycle 10 of operation. This failure resulted in significant fuel loss from one fuel rod to the primary system during operation and additional fuel loss from this rod during handling of the I-24 assembly during the outage. The purposes of this inspection were to 1) evaluate the licensee's efforts to find the missing ~900 g of UO_2 from the failed rod, 2) evaluate the loose parts in the primary coolant, and 3) evaluate the licensee's ability to detect fuel failures during cycle 11 operation.

6.1 Search For Missing Fuel

Higher than normal activities were found on primary system and core components but these activities account for only 6 to 7% of the total fuel lost from assembly I-24. The core was examined to the maximum extent possible without a full core off-load.

The reactor cavity tilt pit was where the failed fuel rod was stripped from the assembly during the fuel handling operations. However, a significant quantity of the fuel may have been lost from the fuel rod prior to its transport to the tilt pit. The bottom of the tilt pit was examined, however, a thorough visual examination was not possible because of hoses and machinery at the bottom of the pit. The licensee intends to drain the pit to a lower water level and see if they can find fuel fragments with activity detectors. The tilt pit was vacuumed and drained after the rod was stripped from the assembly in the tilt pit. A survey of the filters following this first vacuuming of the pit resulted in higher than normal activity levels but only accounted for less than 1% of the lost fuel. The majority of the piping (>90%) that leads from this drain was surveyed and while some small increases in activity were noted they were not at the high levels expected if large fragments of fuel were present. However, the less than 10% of piping not surveyed was located in concrete nearest the drain location. From the piping surveys, and surveys of the filters from three separate vacuuming efforts on the bottom of the tilt pit, less than 1% of the total fuel loss had been accounted for in the tilt pit.

The team concluded that it was not likely that the licensee would find significant additional quantities of the missing fuel. The fuel appeared to be either hiding in inaccessible locations in the primary system, the drains of the reactor cavity tilt pit and the adjacent spent fuel pool, or, more likely, a combination of the above.

6.2 Loose Parts in the Primary Coolant

There are parts missing from the failed fuel assembly that are either in the primary system or in the reactor cavity tilt pit. Among the missing parts are 1) a half diameter piece of cladding approximately 20-inches in length, 2) three to four pieces of the spacer grid including a lantern spring (each is estimated to be less than an inch in length and less than one-half inch in width), and 3) an insulator disk (approximately the size of a fuel pellet). These loose parts have the potential of causing further fuel failures due to debris fretting if they reside in the primary system. They may also contribute to damage of other primary system components. However, historically, debris in the primary system has primarily been a fuel failure problem rather than significantly impacting other components. It should be noted that of 201 fuel assemblies in the cycle 11 core, 136 assemblies are debris resistant assemblies by design.

6.3 Ability to Detect Fuel Failures in Cycle 11

If a significant quantity of the fuel missing from the failed rod (>10%) resides in the reactor coolant system, the ability to detect further fuel failures will be hampered. The licensee had significantly improved their failed fuel detection capabilities for cycle 11 operation. They expanded their activity measurements to include additional radioactive isotopes that help in identifying fuel failures with high tramp (background) activity levels. The licensee has also improved their analytical capabilities. Even with these increased measurements and capabilities the licensee will most likely have difficulty in detecting any small failures from a small number of failed rods; however, they will be able to detect fuel failures with large defect sizes such as those experienced in cycles 9 and 10. These were well below Technical Specification limits.

The licensee had an action plan for addressing increased coolant activity levels. However, this plan left out some of the actions recommended in EPRI report EPRI-NP5521. In addition, the highest activity level that can be achieved before the licensee will consider shut down, or derating, was close to the Technical Specification limit for coolant activity. These action levels were being reevaluated by the licensee.

No violations, deviations, unresolved or inspector followup items were identified in this area.

7.0 AIT Report Findings Compliance Review

A review was conducted to evaluate findings of the NRC Augmented Inspection Team (AIT), as documented in Inspection Report 50-255/93018(DRS), against applicable regulatory requirements. The review identified a number of examples of noncompliance with requirements. These are discussed further below, organized by their applicability to the three broad conclusions reached by the AIT.

7.1 The Licensee's Organization Had a Less than Questioning Attitude

Operating Cycles 10 and 11 involved the use of previously burned fuel assemblies (from the I-series fuel) to provide reactor vessel neutron flux reduction shielding. At the beginning of Cycle 10, the I-series assemblies had operated three full fuel cycles. The licensee had little or no experience operating fuel assemblies beyond three cycles. A review of the proposed extended use of I-series bundles was performed pursuant to 10 CFR 50.59. This review was required to fully consider whether the proposed action involved any Unreviewed Safety Questions (USQ). The conditions of the proposed use involved placing the assemblies at the periphery of the reactor core, where the neutron exposure spectrum (and other parameters) were different from other core locations. In particular, the neutron spectrum at the periphery of the core was proportionally higher in "fast" neutrons and lower in "thermal" neutrons compared to non-periphery locations. A full consideration of the potential that a USQ might be involved required that the effects of the unique neutron spectrum on the I-series assemblies be specifically evaluated.

The licensee's documented 50.59 analyses did not include consideration of the effects of the unique neutron spectrum on the I-series assemblies. Those analyses were therefore incomplete. The written safety evaluation for the subject use of the I-series assemblies did not provide complete bases for the determination that there was no unreviewed safety question. This is an apparent violation of 10 CFR 50.59.(b)(1), which requires such written bases (Violation 50-255/93020-02).

During Cycle 10, it became apparent that fission products and transuranics were present in the primary coolant. The licensee did not have or develop procedures to evaluate this condition with such scrutiny as to correctly identify the root cause. The cause was ascribed to "tramp" uranium/fuel left in the coolant from fuel cladding failures during the previous cycle, Cycle 9. This was not correct.

As a consequence of failure to identify that a low power peripheral fuel rod had failed in service, the fuel was not inspected for damage after Cycle 10. The damage to fuel assembly I-24 was not detected, and the assembly was returned to the reactor for Cycle 11.

Utilization of primary coolant radiochemistry testing procedures which did not assure that reactor fuel was performing properly in service, is an apparent violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control" (Violation 50-255/93020-03a).

7.2 Conservative Decisions were not Made When Warranted

As noted above, the I-series fuel assemblies were not inspected between Cycles 10 and 11. This was despite radio-chemistry evidence of fuel in the primary coolant system and despite the fact they were the oldest assemblies in the reactor. Furthermore, the I-series assemblies were known to have been fabricated with grid straps which were susceptible to relaxation due to strap growth from prolonged exposure to radiation. This potential relaxation would impose a risk of fuel pin vibration and fretting against the grid strap, which could damage the cladding of the pin. This is potentially what occurred.

Failure to verify by test or inspection that I-series fuel assemblies would perform acceptably in service after five previous cycles in the reactor is another apparent violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control" (Violation 50-255/93020-03b).

When the Upper Guide Structure (UGS) was lifted from the reactor on July 6, 1993, and a fuel bundle from core location Z-11 stuck onto the UGS, it marked the third occurrence of this problem. Root cause analyses and corrective actions for the previous events failed to prevent recurrence. Following the event of September 3, 1988, no corrective actions were taken with respect to the alignment pins. After the event recurred on February 29, 1992, the alignment pins at location Z-11 were determined to be bent and they were straightened. Procedure changes were made to provide protection for the alignment pins during handling of the UGS, so they would not become bent again. These changes failed. After the event of July 6, 1993, the alignment pins were found to be bent once more. The pins have now been replaced and a series of actions taken to eliminate more potential causes of pin damage, as well as other potential causes of interference between the UGS and the fuel.

Failure to preclude repetition of interference between core components, which caused fuel mishandling events, is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" (Violation 50-255/93020-04).

7.3 Management Expectations were not Effectively Applied

Controls which were developed for the lift of the Upper Guide Structure (UGS) were not positively and effectively applied. The load cell equipment used for the lift on July 6, 1993, was not the equipment specified by the applicable procedure, No. RVI-M-1, Revision 16, "Removal and storage of the Upper Guide Structure."

The procedure specified a TI-2000 load cell but a J-300 device was actually used. A procedure stipulation (Section 5.3.6.g) to follow Work Order No. 24301781 for steps to use the load cell was violated in that the readout device was not zeroed as required by step 3.3.A.7. In addition, the specified upper load limit of 62,000 pounds (Section 5.3.14) was exceeded by the indicated load of 62,800 pounds after the UGS was raised about six inches. A licensee supervisor was present during this evolution and did not intercede effectively to enforce compliance.

Failures to implement requirements of the UGS lift procedure as described above are apparent examples of violations of Technical Specification 6.8.1.b (Violation 50-255/93020-05a).

The activities associated with recovery of the fuel assembly which stuck to the UGS on July 6, 1993, though successful in safely returning the assembly to the reactor without damage, were not positively controlled in all respects. Specifically, the RWP and procedure FHSO-18, "Recovery of Bundle SAN-8" were not properly implemented. The limitations of procedure FHSO-18 steps 4.2.6 and 5.2.1, in instructing that chainfall tension be limited to a combined load of 1500-1600 pounds, were violated. The chainfalls were tightened to a combined load of 2300 pounds.

Failure to implement requirements of the stuck fuel assembly recovery procedures as described above are an additional apparent example of violation of Technical Specification 6.8.1.b (Violation 50-255/93020-05b).

Seven apparent violations, some involving more than one example, were identified. No deviations, unresolved items, or inspector followup items were identified.

8.0 Public Meeting on September 9, 1993

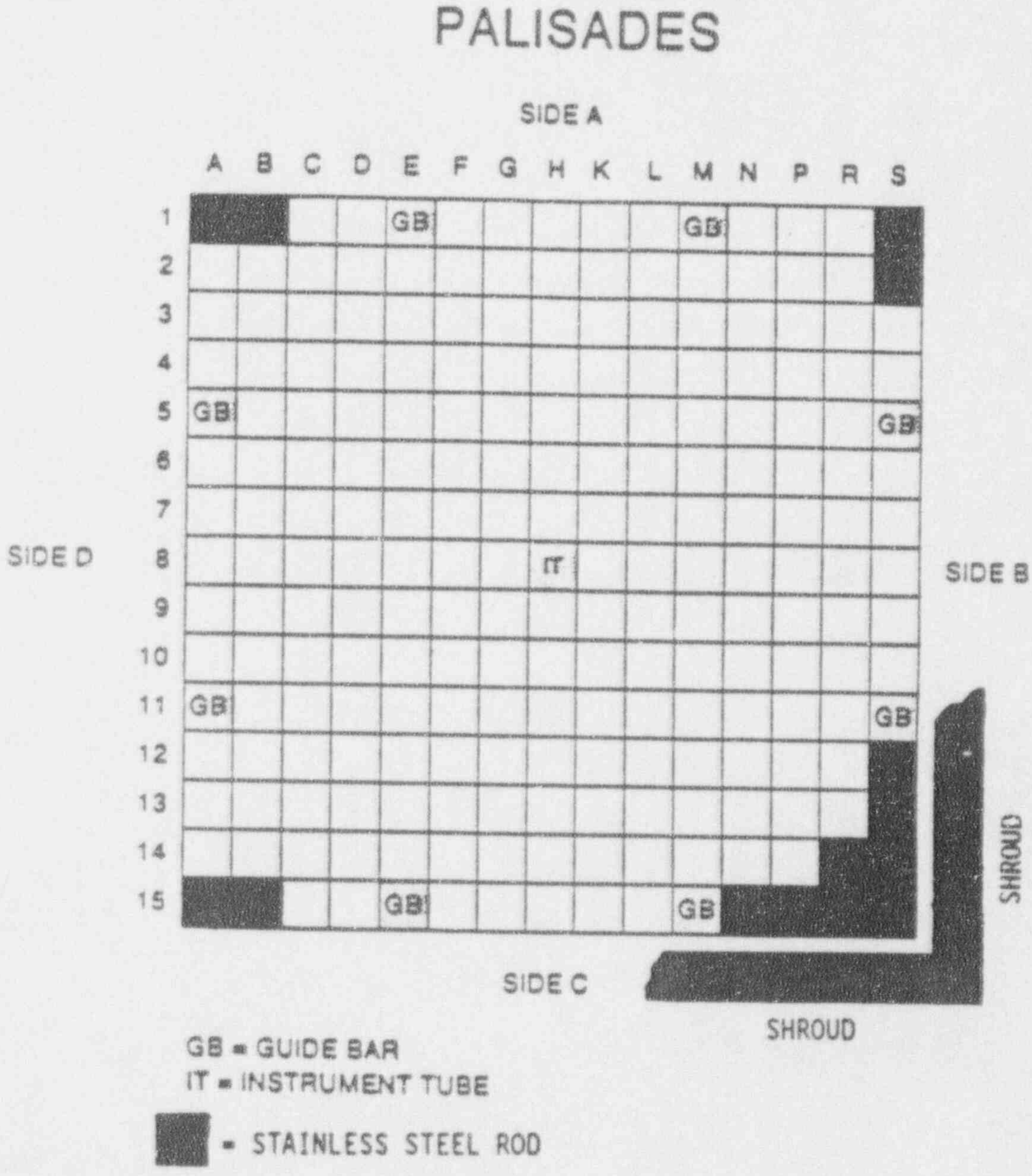
A public meeting was held between managers of Consumers Power Company and the NRC on September 9, 1993. The meeting was held in the Holiday Inn, Route 94, in Benton Harbor, Michigan. The NRC was represented by Hubert J. Miller, Deputy Regional Administrator of Region III and members of his staff, and James G. Partlow, Associate Director for Projects for the Office of Nuclear Reactor Regulation, and members of his staff. Consumers Power Company was represented by David Hoffman, Vice President for Nuclear Operations, Gerald Slade, General Manager of the Palisades Plant, and members of his staff. The purpose of the meeting was for the licensee to present the results of the root cause assessments done for the damaged fuel bundle I-24 and the fuel bundle lifted with the UGS. Mr. Miller opened the meeting, the licensee made the presentation, then Mr. Miller closed the meeting and responded to questions from the public. Licensee representatives also remained to respond to questions.

9.0 Exit Meetings

Interim exit meetings were conducted with licensee representatives on August 21, 1993, and August 27, 1993. A final exit was conducted September 29, 1993. The first exit was by the team of inspectors reviewing the cycle 11 core reload safety evaluation. The second exit interim was by the team which observed routine activities. The final exit summarized the results of the NRC review of the AIT findings for regulatory issues. The inspectors summarized the scope and findings of the inspection. The licensee acknowledged the statements made by the inspectors, The inspectors also discussed the likely informational content of the inspection report with regards to documents and processes reviewed by the inspectors during the inspection and the licensee did not identify any such documents or processes as proprietary.

FUEL ASSEMBLY ROD LOAD MAP

CYCLE 11 L-ASSEMBLY DESIGN

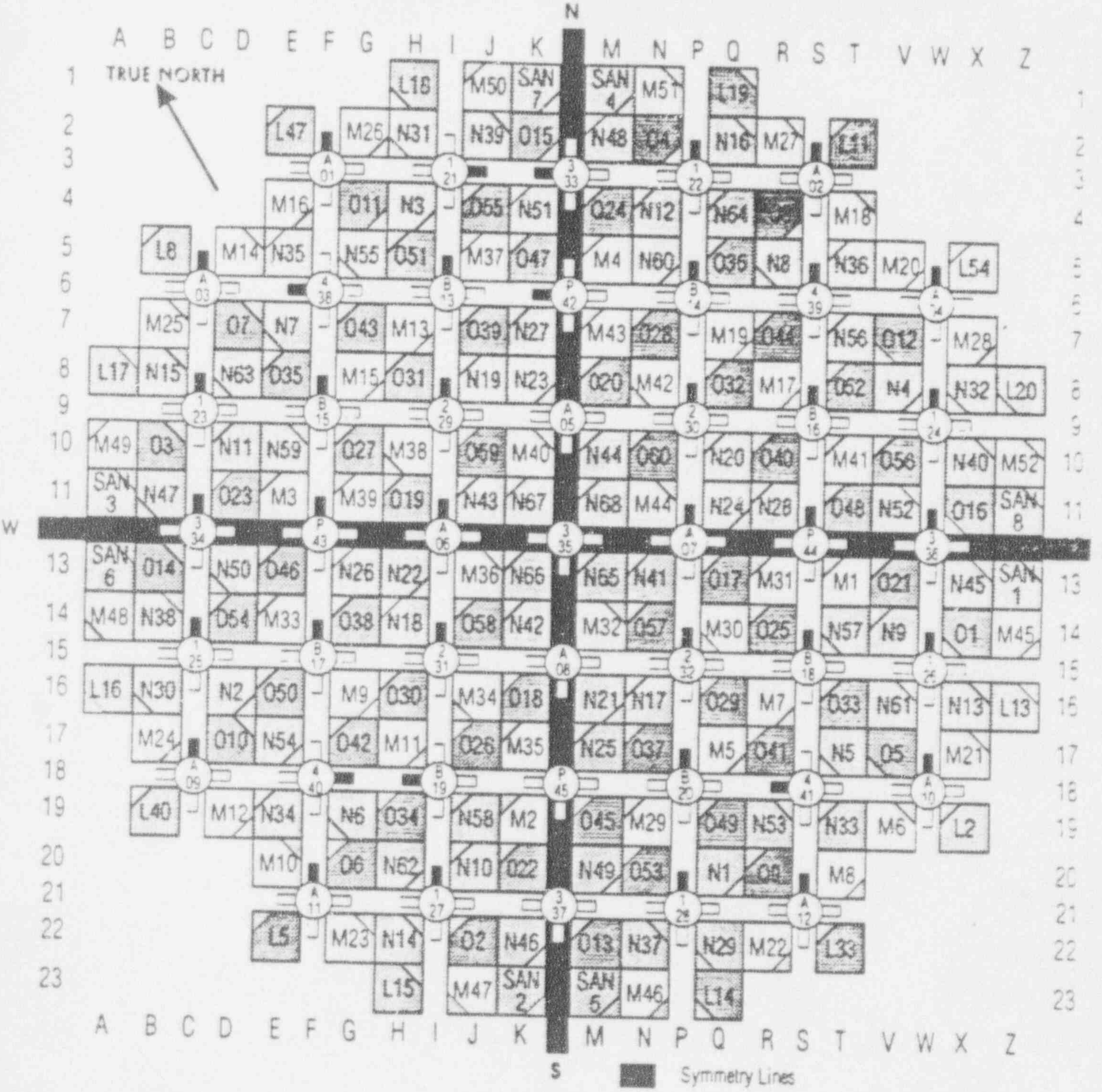


EXAMPLE PATTERN SHOWN FOR THE LOADING OF 14 STAINLESS STEEL RODS INTO AN L ASSEMBLY BASED ON ASSEMBLY TOP VIEW.

SYMMETRIC PATTERN USED FOR ALL 16 L ASSEMBLIES WITH THE 8 STAINLESS STEEL ROD CORNER ALWAYS POSITIONED TO BE AT THE CORE SHROUD CORNER.

Figure 1 - Revised L-assembly for Cycle 11

Revised
CYCLE 11 CORE PLAN
PALISADES NUCLEAR PLANT



Batch O: New Fuel   Batch N: Once Burnt   SAN: Once Burnt   Batch M: Twice Burnt   Batch L: Three Times Burnt

SAN assemblies contain Stainless Steel pins

Cycle 11 Core Plan uses ¼ core rotational symmetry

Figure 2 - Revised Cycle 11 Core Design