ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-528 50-529 50-530
License Nos.:	NPF-41 NPF-51 NPF-74
Report No.:	50-528/97-11 50-529/97-11 50-530/97-11
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Road Tonopah, Arizona
Dates:	March 31 through April 4, 1997
Inspector:	John E. Whittemore, Reactor Inspector
Approved By:	Dr. Dale A. Powers, Chief, Maintenance Branch Division of Reactor Safety

Attachment: Supplemental Information

EXECUTIVE SUMMARY

Palo Verde Nuclear Generating Station, Units 1, 2, and 3
NRC Inspection Report 50-528;-529;-530/97-11

This inspection consisted of a review of the licensee's planned and implemented corrective action to address previous inspection findings and licensee event reports. The inspection report covers a 1-week period on site by one region-based inspector.

Maintenance

- The licensee's corrective actions were adequate to permit closure of several inspection open items (Section M8).
- The failure to meet the limiting conditions for operation in Unit 3 for planar radial peaking factors was a noncited violation of Technical Specification 3.2.2 (Section M8.1).

Engineering

- The licensee's corrective actions were adequate to permit closure of several inspection open items (Section E8).
- The failure to establish measures to assure translation of the design basis for reactor core reloads into the Technical Specifications was a noncited violation of Criterion III of Appendix B to 10 CFR 50 (Section E8.3).

Report Details

Summary of Plant Status

During the inspection, Units 1 and 2 were operating at or near full power. Unit 3 was starting up after a refueling outage.

II. Maintenance

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Licensee Event Report 50-530/94008: Inadvertent cross-wiring of incore detectors during refueling outage recovery.

While Unit 3 was shutdown in Mode 5, the licensee discovered that it had operated at power above 20 percent from June 21 through November 26, 1994, with the core operating limit supervisory system input signals supplied from temporarily installed mini-incore instruments, instead of the standard-incore instruments. Cross wiring the standard and mini-incore instruments resulted in a violation of Technical Specification 3.2.2, "Planar Radial Peaking Factors," in that the measured planar radial peaking factors were greater than the factors used in the core operating limit supervisory system and the core protection calculators during power operation above 20 percent.

Prior to the event, in 1991, eight mini-incore instruments were installed in Unit 3 under an approved temporary modification to determine if mini instruments could be used to eventually replace standard instruments. The mini instruments fit inside the movable calibration tubes of the standard instruments. The mini instruments were intended to feed a temporary data acquisition system and be connected to this system at the seal table. The connectors were identical on both types of instruments and the mini instruments were inadvertently connected to the core operating limit supervisory system and the core protection calculators following the fourth refueling outage in 1994.

The inspector assessed the efforts to correct this condition and preclude recurrence of the event. Although operation occurred outside Technical Specification 3.2.2, licensee personnel determined that there was no adverse safety impact due to the small difference in the output of the mini and standard incore instruments. Licensee personnel determined the root cause of the event to be personnel error by the refueling support, and the instrument and control groups. The cause determination revealed that these personnel failed to follow appropriate work instructions dealing with temporary modifications.

The initial corrective actions were to reconnect the instruments correctly and issue an "I&C NEWSFLASH," to alert personnel to the requirements for implementing and maintaining temporary modifications. As a followup action, an engineering evaluation of the temporary modification control process was performed. This evaluation identified several temporary modification process control actions that would have prevented the event had the actions been followed. Therefore, no revisions were required for the temporary modification process. According to records provided, the temporary modification was removed on October 18, 1995. This item is closed based on removal of the temporary modification and successful completion of the planned corrective action.

Failure to operate within the limiting conditions for operation for planar radial peaking factor, was a violation of Technical Specification 3.2.2. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-530/9711-01).

M8.2 (Closed) Violation 50-529/96006-01: Inadequate procedural measures to maintain the design basis of the Unit 2 upper guide structure.

The Unit 2 upper guide structure was damaged during placement on and/or retrieval from its storage stand in the containment refueling cavity. The damaged upper guide structure subsequently damaged the fuel assembly located in the reactor core position A07 during reassembly of upper reactor vessel components after fuel loading. The immediate action consisted of removing the stuck fuel assembly and performing repairs to the upper guide structure.

The licensee performed a detailed investigation and determined that weak procedural controls, incorrect assumptions made by personnel moving the upper guide structure, and a less than optimum design of the upper guide structure alignment system, contributed to the root cause of the event.

The inspector reviewed documentation and verified that procedures for moving the upper guide structure had been revised to require extensive use of underwater video cameras to aid in placing and removing the upper guide structure from its storage stand. Additionally, documentation indicated the administration of extensive retraining of nuclear fuel and reactor vessel component handling personnel. Finally, the licensee initiated and commenced modifying the alignment system for placing the upper guide structure on its storage stand. At the time of the inspection, the modification had been installed in Unit 3, and consisted of two permanently-installed guide pins positioned adjacent to the storage stand. The pins, designed to engage the lift rig bushings, were machined with a short taper that expanded to an outer diameter 1/2 inch less than the upper guide structure lift rig guide bushings' inner diameter.

The licensee had originally committed to implement the alignment system design change during the next refueling outages (e.g., first opportunities) for the three units. These modifications would have been completed during the Unit 2 outage that would have ended in November 1997. However, during the fall of 1996 outage for Unit 1, a handling event occurred that resulted in deforming one of the lift rig outriggers which held one of the guide bushings. It was not possible to implement the guide pin modification until the outrigger issue was resolved. The resolution of this new problem resulted in the development of a new modification to strengthen the outrigger assembly. Therefore, the guide pin modification was not implemented as scheduled in Unit 1 and neither was the modification to strengthen the guide bushing outrigger. Both of the modifications were scheduled to be installed in Unit 1 during the spring of 1998 outage. Both modifications were successfully installed in Unit 3 during the most recent outage and were scheduled to be installed in Unit 2 during the fall of 1997 outage. The inspector queried the licensee's representative about whether the original commitment to the NRC in the response to a notice of violation had been formally revised. Subsequently, on April 3, 1997, the licensee issued a letter to the NRC changing the commitment date to March 1998 when the modification would be installed in Unit 1.

This item is closed on the basis of the successful modification installation of the upper guide structure alignment system in Unit 3 and a licensee commitment to install the modification in Units 1 and 2 by March 1998.

M8.3 (Closed) Unresolved Item 50-528;-529;-530/96009-01: Unclear guidance and nonspecific performance criteria for monitoring the performance of structures as required by the Maintenance Rule.

The Maintenance Rule Baseline Team Inspection identified the use of design basis information instead of specific performance criteria for the monitoring of structures and the unclear guidance for ensuring that structures would be moved to Category (a)(1) when required, as significant weaknesses in the implementation of the licensee's Maintenance Rule Program.

In response to this finding, the licensee initiated a programmatic change through the issue of new Procedure 81IG-0ZZ01, "Civil Monitoring Process," and revision of the following procedures:

- Procedure 30DP-0MR01, "Maintenance Rule"
- Procedure 81DP-0ZZ01, "Civil SSCs Monitoring Program"
- Procedure 711G-0EP02, "(a)(1)/(a)(2) Evaluation and Goal Setting"
- Procedure 70DP-0EM01, "Risk Management Expert Panel"

The inspector reviewed the new procedure and the changes to the existing procedures. The integrated effect of all changes was to provide clear structure performance criteria and a process for placing the structures in Category (a)(1) or

back to Category (a)(2). There were no changes to the categorization of structures, as a result of the change in performance criteria for structures. Therefore, this item is resolved and closed.

M8.4 (Open) Inspection Followup Item 50-528/96019-01: Ultrasonic testing calibration block was constructed from material that was different from the material that was examined.

Scheduled inservice inspection ultrasonic examinations were not performed because the licensee could not determine what calibration blocks to use with the specific piping weld. The confusion resulted from different blocks required by the "zone drawing summary" and the "outage inservice plan table." However, neither block was constructed from the same material as the piping. This was a condition which did not conform to the ASME Code.

At the time of the inspection, licensee personnel had not determined the cause of the event. Some of the corrective action that would obviously be required had been initiated. A new calibration block had been manufactured and certified. Also, licensee personnel had addressed the generic implications by identifying similar configurations in the three units that were affected. Some re-examination had been performed and accepted.

This inspection followup item remains open pending the determination of the root cause, identification of the scope of the required corrective action, and the planned implementation of the corrective action.

III. Engineering

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Violation 50-528;-529;-530/94035-01: Failure to initiate a timely formal corrective action document, in accordance with the corrective action program, when a condition was identified that required the implementation of additional restrictions to satisfy design basis safety analysis assumptions.

Licensee personnel involved in core reload analyses, identified conditions that required: (1) implementation of a reactor coolant boron concentration restriction, and (2) required core protection calculator operability in Modes 3, 4, and 5. These conditions were identified and discussed by licensee personnel in November 1993 and February 1994. The additional restrictions were required to maintain design basis safety analysis assumptions. A condition report/disposition request was not initiated when the conditions were identified.

Licensee personnel identified the root cause of this violation as a misunderstanding of the corrective action program by individuals involved in nuclear fuel management activities. The individuals involved had inappropriately concluded that resolution of concerns was part of the normal work process and, therefore, initiation of a corrective action document was not necessary.

The immediate corrective action was to initiate a corrective action document to address the issue of the untimely corrective action process initiation and assign the nuclear assurance department to conduct a briefing for all personnel in the nuclear fuel management group. The briefing consisted of identifying the requirements for initiating a condition report/disposition request as required by the licensee's corrective action process. As followup corrective action, the training department developed a 2-hour training course on the corrective action program that identified and reinforced when implementation of the corrective action process was required. This course was administered to 270 management and supervisory employees. Managers and supervisors then provided briefings to all front line employees. The followup corrective action was completed prior to April 1, 1995. Therefore, this violation is closed on the basis of the fully implemented, satisfactory, corrective action.

E8.2 (Closed) Inspection Followup Item 50-528;-529;-530/94035-02: Questionable adequacy of initial and continuing training program for personnel in various engineering groups with core reload analysis responsibilities.

The licensee had purchased a training package from the nuclear fuel vendor to train Arizona Public Service engineering personnel to perform core reload analyses and design. During the original inspection, due to the lack of documentation, the NRC inspectors were unable to determine the extent of initial and continuing training administered to licensee personnel with responsibility for core reload analyses and design.

In response to this issue, the licensee developed a more detailed training program for engineering personnel assigned responsibilities in the nuclear fuel management area. The "Training Program Description for Engineering Personnel" was revised to require the development and implementation of "Job Qualification Guides" for nuclear fuel management personnel. Guides were in place for personnel assigned to the fuel management engineering areas of nuclear analyses, transient analyses, reload analyses, and fuel cycle services. There was also a computer software development and control job qualification guide that was common to all fuel management disciplines. However, only selected personnel were permitted to qualify for developing and controlling software. These guides identified the skills and knowledge that an assignee must perform or demonstrate to be designated as qualified to perform engineering tasks in the specific nuclear fuel management area and to develop and control software. A matrix was provided to the inspector which identified the current training and qualifications of all personnel in the nuclear fuel management organization.

Continuing training was provided to nuclear fuel management personnel in accordance with the "Training Program Description for Engineering Personnel." Nuclear fuel management section leaders were required to select training items identified for the quarterly industry events training and provide this information in detailed form to personnel in their sections. Continuing training administered to fuel management personnel was documented in files maintained by the group supervisors. This followup item is closed on the basis that the inspector was able to determine that the appropriate personnel had been trained to perform specific tasks, and that the training was adequate.

E8.3 (Closed) Licensee Event Report 50-528;-529;-530/95002: Identification of conditions with the potential to place the units in an unanalyzed condition.

During activities associated with baselining conditions, requirements, and assumptions to be used in the core reload analysis process, licensee personnel identified the potential for an unanalyzed condition. The Technical Specification Limiting Conditions for Operation 3.3.2, "Engineered Safety Features Actuation System Instrumentation," and 3.5.2, "Emergency Core Cooling Subsystems," were inconsistent with the assumptions used in the steam line break analysis for validation of the temperature-dependent shutdown margin, while in Mode 3 above 500 degrees Fahrenheit. The conditions affected were related to the safety injection actuation setpoint and required operable emergency core cooling trains, respectively. Additionally, the analysis only considered the shutdown margin specified in Limiting Condition for Operation 3.1.1.2, "Shutdown Margin - Reactor Trip Breakers Closed," for any rod withdrawn. However, Technical Specification 3.1.1.1, "Shutdown Margin - Reactor Trip Breakers Open," (all rods in) was applicable and actually more restrictive. The licensee determined that although this oversight would have reduced the shutdown margin, it would not have resulted in the plant exceeding any safety limit during the design basis accident.

The immediate corrective action was to establish administrative controls to apply Technical Specification 3.5.2 any time reactor coolant T_{cold} was greater than or equal to 500 degrees Fahrenheit. This action ensured that pressurizer pressure was maintained greater than 1700 psia when in Mode 3. The licensee also required Technical Specification 3.1.1.2 to be applicable for any unit in Mode 3.

Licensee personnel coordinated with the nuclear fuel vendor to determine the root cause and corrective action for the event. An initial determination was made that Technical Specification 3.1.1.1, "All Rods In Shutdown Margin," was less limiting than Technical Specification 3.1.1.2, "Temperature Dependent Shutdown Margin." However, this was not supported in the analysis documentation. A statement made in a Cycle 1 reload safety analysis document was the apparent basis for the erroneous conclusion. The root cause of the event was determined to be a lack of coordination and the unclear division of responsibilities between groups within the vendor's organization. A vendor letter agreed with this conclusion.

Licensee fuel management personnel worked with the nuclear fuel vendor to develop the corrective actions to ensure that the limiting conditions for operation supported the assumptions in the analyses and calculations. The vendor supplied the analyses and calculations to support the revised and subsequently amended Technical Specification setpoints and limits. The revisions were made for all three units.

The inspector reviewed the Technical Specification amendment and the changes to the Updated Final Safety Analysis Report, Chapter 15. It was verified that the license conditions related to the applicable plant parameters, required equipment operability, and engineered safety features actuation setpoint had been changed as necessary. Also, the Updated Final Safety Analysis Report accurately stated the required plant conditions to adequately support the assumptions used in the analysis.

The inspector also verified that the operating procedures for startup and shutdown were revised to administratively control reactor coolant system temperature and pressure, as well as, required equipment operable, while in Mode 3 to preserve the assumptions used in the analysis for adequate shutdown margin during a steam line break event. Therefore, this licensee event report is closed based on the completion of adequate corrective action.

The failure to establish measures to assure that the updated Final Safety Analysis Report design basis was correctly translated into the Technical Specifications was a violation of Criterion III to Appendix B of 10 CFR 50. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-530/97011-02).

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on April 4, 1997, and in a subsequent telephonic discussions on April 15 and 29, 1997. The licensee acknowledged the findings presented. The licensee did not identify as proprietary any information or materials examined during the inspection.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. Cawley, Director, Nuclear Fuel Management
R. Hazelwood, Engineer, Nuclear Regulatory Affairs
J. Hesser, Director, Engineering
W. Ide, Vice-President, Engineering
D. Mauldin, Director, Maintenance
M. Radspinner, Section Leader, Mechanical Design Engineering
M. Reid, Section Leader, Nuclear Fuel Management
S. Ryan, Section Leader, Maintenance Support

NRC

D. Garcia, Resident Inspector

INSPECTION PROCEDURES USED

IP 92902	Followup-Maintenance
IP 92903	Followup-Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-530/9711-01	NCV	Violation of Technical Specification 3.2.2
50-528/-529/ -530/9711-02	NCV	Violation of Criterion III of Appendix B to 10 CFR 50

Closed

50-530/94008	LER	Technical Specification Violation Due To Cross-Wired In-Core Instruments
50-528;-529; -530/94035-01	VIO	Failure To Initiate A Formal Corrective Action Document

50-528;-529; -530/94035-02	IFI	Questionable Adequacy of Training Program For Core Reload Analysis Personnel
50-528;-529; -530/95002	LER	Potential To Place The Units In An Unanalyzed Condition
50-530/96006-01	VIO	Inadequate Procedural Measures To Maintain The Design Basis Of The Unit 2 Upper Guide Structure
50-528;-529; -530/96009-01	URI	Unclear Guidance And Performance Criteria For Monitoring Structures Under The Maintenance Rule Program
50-530/9711-01	NCV	Violation of Technical Specification 3.2.2
50-528/-529 -530/97011-02	NCV	Violation of Criterion III of Appendix B to 10 CFR 50

Discussed

50-528/96019-01	IFI	UT Calibration Block Material Different From Piping Material