ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-528/96-01
50-529/96-01
50-530/96-01

Licenses: NPF-41
NPF-51
NPF-74

Licensee: Arizona Public Service Company
P.O. Box 53999
Phoenix, Arizona

Facility Name: Palo Verde Nuclear Generating Station, Units 1, 2, and 3

Inspection At: Maricopa County, Arizona

Inspection Conducted: March 4-8 and April 22-26, 1996

Inspectors: Linda Joy Smith, Reactor Inspector, Engineering Branch
Division of Reactor Safety

Chris Myers, Reactor Inspector, Engineering Branch
Division of Reactor Safety

Approved: Chris A. VanDenburgh, Chief, Engineering Branch
Division of Reactor Safety

Date: 5/29/96

Inspection Summary

Areas Inspected (Units 1, 2, and 3): Routine, announced inspection of the licensee's self-assessment effort related to engineering and corrective action.

Results (Units 1, 2, and 3):

- The inspectors determined that a qualified self-assessment team conducted an independent and objective assessment of the licensee's engineering and corrective action programs (Section 1.1.2).

- The inspectors found the scope and depth of the self assessment to be ambitious and sufficient to satisfy all the inspection requirements of NRC Inspection Procedures 37550, "Engineering," and 40500, "Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems" (Sections 2.1.2 and 2.2.2).

- The inspectors concluded that, with some minor exceptions, the self-assessment team had appropriately identified and dispositioned problem areas and potential weaknesses. Neither the licensee's self-assessment team nor the inspectors identified inoperable equipment (Sections 2.3.2 and 4.2.6).

- The self-assessment team concluded that the material condition of the auxiliary feedwater, diesel generator, and selected important-to-safety systems was generally good and that these systems were fully capable of performing their intended safety functions (Section 3.2.1).

- The self-assessment team noted that engineering management had focused on prioritizing the workload and reducing the engineering backlog. They determined that equipment issues affecting system reliability were being dealt with effectively (Section 3.2.1).

- The self-assessment team found that the licensee had formed engineering teams, led by system engineering personnel, which were actively maintaining and improving system performance (Section 3.2.3).

- The self-assessment team found that licensee personnel effectively used probabilistic risk assessment information for decision making and prioritization. However, in two cases (one identified by the self-assessment team and one identified by the NRC), licensee personnel did not conservatively address risk implications (Sections 3.2.4 and 4.2.2).

- The self-assessment team found that licensee personnel were effectively maintaining a conservative design basis for the plant. The self-assessment team concluded that engineering activities were improving (Section 3.2.5)

- The inspectors identified one case where licensee personnel had not consistently translated the licensing basis for the nonessential train of the auxiliary feedwater system into the design basis for the plant (Section 4.2.3).

- The inspectors noted that the licensee had not performed a design-basis verification for the condensate transfer system, which included the condensate storage tank and the auxiliary feedwater mini-flow lines (Section 4.2.4).

- The self-assessment team found that licensee personnel performed engineering calculations, evaluations, and dispositions with satisfactory rigor and technical accuracy (Section 3.2.5).

- The self-assessment team concluded that the new plant modification program was working well. However, they found that some older plant modification program issues still existed, such as the need to improve control of "abandoned-in-place" modifications (Section 3.2.6).

- The self-assessment team concluded that engineering personnel effectively provided technical direction and input to help the plant personnel resolve significant issues. However, the self-assessment team found that engineering personnel did not always effectively deal with emerging technical issues which were determined to be of lesser significance (Section 3.2.7).

- The self-assessment team observed that management oversight, particularly through the large, process-oriented self-assessments, Nuclear Assurance audits, and Independent Safety Evaluation assessments had been rigorous and critical for both the design modification and the corrective action process (Section 3.2.8).

- The self-assessment team concluded that, in general, problems were being identified, evaluated, and resolved. They found that the licensee's ability to effectively resolve issues and prevent recurrence of significant conditions adverse to quality had improved (Section 3.2.9).

- The self-assessment team found a general reluctance to write condition report/disposition requests (Section 3.2.9).

- The self-assessment team concluded that implementation of the recently enhanced operability determination process was weak. The team identified cases where operability determinations were not completed in a timely manner (Section 3.2.9)

- The inspectors identified additional examples of operability determinations which were not performed as recommended by Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections and Resolution of Degraded and Nonconforming Conditions and On Operability." While not a requirement, the licensee stated that it was their policy to implement Generic Letter 91-18 (Section 4.2.5).

- The self-assessment team noted that specific problems identified on condition report/disposition requests were generally corrected but repetitive and/or related problems were not always thoroughly analyzed to determine if more extensive evaluation or corrective action was needed (Section 3.2.9).

Summary of Inspection Findings:

- Two non-cited violations were identified (Section 3.2.9).
- Inspection Followup Item 50-528/9601-01: 50-529/9601-01: 50-530/9601-01 was opened (Section 4.2.3).

TABLE OF CONTENTS

ATTACHMENTS:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Team Member Credentials

DETAILS

1 TEAM COMPOSITION (40501)

1.1 Qualifications, Objectivity, and Independence

1.1.1 Inspection Scope

The purpose of this inspection was to determine the effectiveness of the licensee's self assessment of their engineering and corrective action programs. In letters, dated December 12, 1995, and January 19, 1996, the licensee proposed to perform a self-assessment of their engineering and corrective action programs in accordance with the guidance of NRC Inspection Procedure 40501, "Licensee Self Assessments Related to Team Inspections." The option of permitting licensees to conduct a self assessment in lieu of planned NRC team inspection is an NRC program aimed at minimizing regulatory impact and utilizing NRC resources more efficiently. Region IV NRC team inspections were planned to accomplish the core inspection program requirements of NRC Inspection Procedures 37550, "Engineering" and 40500, "Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems."

The inspectors reviewed the qualifications, objectivity and independence of the personnel performing the self assessment.

1.1.2 Observations and Findings

The letters referenced above included a description of the qualifications of the team members. The inspectors reviewed the qualifications of the team members and found they exhibited a wide scope of engineering disciplines. Each member possessed significant engineering experience. Subsequently, the licensee added one additional member to the self-assessment team. A description of his credentials, which are also acceptable, is attached to this report.

The inspectors noted that the self-assessment team was primarily staffed with personnel from the Palo Verde Nuclear Generating Station. To provide an independent perspective, the licensee included two consultants and two engineers on loan from other facilities as team members. The NRC accepted the credentials and experience of the assessment team in a memo to William L. Stewart, Executive Vice President, Nuclear, Arizona Public Service Company, dated February 15, 1996.

The inspectors noted that the self-assessment team questioned the effectiveness of several programs which minimally met regulatory requirements. As a result of questions from the self-assessment team, the licensee planned program upgrades in many areas. In a few cases the self-assessment team did

not identify all of the issues associated with their findings because of their familiarity with current plant practices. However, the inspectors concluded that a qualified self-assessment team conducted an independent and objective assessment of engineering and corrective action activities.

2 LICENSEE SELF-ASSESSMENT PROCESS (40501)

2.1 Scope

2.1.1 Inspection Scope

In the letters referenced above, the licensee provided the NRC their engineering and corrective action self-assessment plan. The inspectors compared the submitted inspection plan against the requirements of NRC Inspection Procedures 37550 and 40500. The inspectors observed in-process assessment activities and interviewed licensee personnel.

2.1.2 Observations and Findings

The inspectors determined that the licensee's assessment plan included all the key elements listed in NRC Inspection Procedures 37550 and 40500. The inspectors found that the self-assessment team selected two safety-related systems for evaluation: auxiliary feedwater and the emergency diesel generator system. These systems were selected based on their contribution to core damage frequency as identified in the Palo Verde Nuclear Generating Station individual plant examination. The team also evaluated engineering and corrective action activities for other important-to-safety systems listed in the referenced NRC inspection procedures.

The self-assessment team examined engineering activities as they related to maintaining the design basis and improving system performance. They evaluated temporary and permanent modifications to ensure compliance with design basis documents. The self-assessment team conducted system walkdowns and reviewed past operating and maintenance history to assess system reliability. The team also reviewed corrective action documents, operating experience review documents and reports of oversight committee activities to assess the effectiveness of licensee controls for identifying, resolving and preventing problems related to these systems.

The inspectors concluded that the scope of the self assessment was sufficient to satisfy the requirements of NRC Inspection Procedures 37550 and 40500.

2.2 Depth

2.2.1 Inspection Scope

The inspectors reviewed the compilation of the self-assessment team's requests for information and the licensee's response to the self-assessment team's questions. The inspectors also reviewed the team's completed checklists, the issued self-assessment report, and the resulting condition report/disposition requests.

2.2.2 Observations and Findings

The self-assessment team developed detailed audit checklists to implement the assessment plan, which had been provided to the NRC. Each team member was designated responsibility for completing specified checklists to describe their findings. The self-assessment team leader used the completed checklists to develop the self-assessment report. The inspectors noted that the self-assessment team's requests for information were appropriately focused on issues with potential nuclear safety impact. The questions were similar to the types of questions which would have been posed by NRC personnel inspecting the same subject area.

The licensee committed significant resources to this effort (i.e., well in excess of the number of core inspection hours planned by the NRC for similar activities). The 12-person, self-assessment team reviewed licensee activities for 3 weeks, resulting in approximately 36 person-weeks of inspection.

The number of requests for information generated by the self-assessment team also provided a qualitative measure of the scope of the licensee's review effort. The team made 146 requests for information, which resulted in the initiation of 26 condition report/disposition requests. The licensee uses condition report/disposition requests to evaluate improvement areas, as well as to identify adverse conditions. Of the 26 condition report/disposition requests, 17 were of sufficient significance to require response by a line organization. The inspectors noted that the self-assessment team identified many possible enhancements, which exceeded regulatory requirements.

The inspectors found the self assessment to be ambitious and of sufficient depth to satisfy the inspection requirements of NRC Inspection Procedures 37550 and 40500.

2.3 Plan and Implementation

2.3.1 Inspection Scope

Two inspectors reviewed the self-assessment team's effort from March 4 through April 26, 1996, in accordance with NRC Inspection Procedure 40501. The inspectors observed the performance of the self-assessment team during the first week of onsite inspection, March 4-8, 1996. The inspectors performed a second week of onsite independent inspection, April 22-26, 1996, to ensure the satisfactory completion of the team's self-assessment. The inspectors performed in-office review of the self-assessment team's findings during the interim weeks.

The inspectors observed the self-assessment team perform system walk downs, interview personnel, and conduct team meetings.

2.3.2 Observations and Findings

The inspectors concluded that the team appropriately identified problem areas and potential weaknesses. The inspectors concurred with the self-assessment team's disposition of the identified issues with some minor exceptions discussed below. The self-assessment team did not identify any examples of inoperable equipment.

3 SIGNIFICANT SELF-ASSESSMENT TEAM CONCLUSIONS (40501)

3.1 Inspection Scope

The inspectors reviewed the self-assessment report, which included three main sections: System Reviews; Engineering; and Ability to Identify, Evaluate and Resolve Problems. The inspectors summarized the licensee's conclusions from each section and the information that the team highlighted in the executive summary.

3.2 Observations and Findings

3.2.1 Material Condition

The self-assessment team concluded that the material condition of the auxiliary feedwater, the diesel generator, and selected important-to-safety systems was generally good and that these systems were fully capable of performing their intended safety functions. These systems were installed in accordance with the design and licensing basis of the plant. The team's conclusion was based on extensive walkdowns. The team took notes of their observations and minor deficiencies were passed to the licensee for action.

3.2.2 Engineering Work Backlog

The self-assessment team noted that engineering management had focused on prioritizing the workload and reducing the engineering backlog. The team also noted that lingering equipment issues were being addressed. For example, the number of temporary modifications and installed drip catches had been reduced. The team also found that the auxiliary feedwater and emergency diesel generator systems' performance had improved. They determined that equipment issues affecting system reliability were being prioritized effectively.

3.2.3 System Engineering

The self-assessment team found that the licensee had formed engineering teams, led by system engineering personnel, which were actively maintaining and improving system performance.

3.2.4 Use of Probabilistic Risk Assessment Information

For the most part, the team found that licensee personnel were effectively using probabilistic risk assessment information for decision making and work prioritization. As an exception, the self-assessment team identified one case where risk implications were not conservatively addressed.

Specifically, the self-assessment team identified that, during a 10 CFR 50.59 evaluation of a proposed modification to the auxiliary feedwater pump turbine steam supply system, licensee personnel had incorrectly determined that a modification, which resulted in an increase in core damage frequency, was acceptable because the core damage frequency increase was small. The self-assessment team noted the 10 CFR 50.59 evaluation was inconsistent with guidance the NRC had previously provided to another licensee (Virginia Power). The NRC had stated that the requirements in 10 CFR 50.59 do not include a specific threshold below which the effects of a core damage frequency change were considered to be inconsequential, and the NRC staff had not endorsed a threshold value below which the effects of a positive core damage frequency change were considered inconsequential.

Licensee personnel reperformed the probabilistic risk analysis with more precise input assumptions and found that the core damage frequency did not increase. As a result, the conclusion from the original 10 CFR 50.59 evaluation remained unchanged; however, the self-assessment team determined that the procedural guidance for performing 10 CFR 50.59 evaluations was not thorough with respect to the proper use of probabilistic risk assessment information. The licensee planned an upgrade to the 10 CFR 50.59 procedure to provide better guidance to the evaluators in this area.

3.2.5 Design Basis Maintenance

The self-assessment team found that engineering personnel were effectively maintaining a conservative design basis for the plant. The team determined that the design basis validation project for the auxiliary feedwater system successfully identified and corrected many deficiencies between the Updated Final Safety Analysis Report and the design basis documents, which were within the scope of the project. The self-assessment team found that plant personnel had accurately reflected the design basis in the design output and configuration documents with a few minor exceptions.

The self-assessment team found that engineering calculations, evaluations, and dispositions were generally performed with satisfactory rigor and technical accuracy. As an exception, the self-assessment team identified two cases where engineering provided nonconservative technical input to shift supervisors to use to determine equipment operability. In one case an operability determination was performed to evaluate the operability of the essential auxiliary feedwater pumps when the associated water-tight doors were inoperable. The operability determination relied on operator compensatory actions, which were not consistent with the assumptions of the design basis flooding calculations. However, the water-tight doors were operable at the time the inadequacy in the operability determination was discovered. The second case involved an operability determination to establish limits for the amount of insulation which could be removed from various safety systems without exceeding the cooling capacity in the pump rooms. These limits were out-of-date and nonconservative for the low pressure safety injection system, one train of containment spray and for the auxiliary feedwater system.

However, the amount of insulation actually removed in the pump rooms was less than the minimum required (based on updated, corrected values). The operability status of plant equipment was unchanged as a result of discovering these inadequacies.

3.2.6 Plant Modifications

The self-assessment team concluded that the new plant modification program was working well. They found that some older plant modification program issues still existed, such as control of "abandoned-in-place" modifications. The team found that in some cases systems had been effectively abandoned-in-place without completing modifications to actually remove the installed equipment. The team concluded that this practice resulted in weak configuration controls.

3.2.7 Technical Support

The self-assessment team concluded that engineering personnel were effectively providing technical direction and input to help the plant resolve significant issues. However, the team found that engineering personnel were not always effectively dealing with emerging technical issues, which were viewed by the licensee to be of lesser significance.

3.2.8 Self Assessments

The self-assessment team observed that management oversight, particularly through the large, process-oriented self-assessments, nuclear assurance audits, and independent safety evaluation assessments had been rigorous and critical for both the design modification and the corrective action process. The self-assessment team found that problems associated with the design modification process had been self-identified during a previous self assessment and corrective action plans were in process. The self-assessment team found that corrective action program weaknesses, which were self-identified in 1994, had been systematically dealt with by plant management. As a result, the corrective action process had been simplified to provide better focus to significant issues. The self-assessments performed in 1995, by both the line organizations and the nuclear assurance department, also resulted in development of corrective action plans to address identified weaknesses and improved performance. The self-assessment team viewed the licensee's commitment to developing a self-assessment culture as a strength.

3.2.9 Problem Identification

In general, the self-assessment team concluded that problems were being identified, evaluated, and resolved. They found that the licensee's ability to effectively resolve issues and prevent recurrence of significant conditions adverse to quality had improved. However, the self-assessment team found a general reluctance to write condition report/disposition requests. The team noted cases where plant personnel identified apparent conditions adverse to quality and failed to document these conditions using the condition report/disposition request process.

Personnel involved in three different assessment processes failed to document their findings on condition reports/disposition requests. As an example, operations personnel identified eight adverse conditions related to tagging and clearances during an operations self assessment without issuing a condition report/disposition request. To address this issue licensee personnel provided additional training for site personnel to ensure understanding of the need to initiate a corrective action document. They implemented a media campaign to stress the use of the corrective action program. Licensee personnel also developed a long term action plan to research and address the cause of the reluctance to write condition report/disposition requests. The failure to identify conditions adverse to quality is a violation of Criterion XVI of 10 CFR Part 50, Appendix B. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

The self-assessment team also concluded that implementation of the recently enhanced operability determination process was weak. The team identified cases where operability determinations were not completed in a timely manner. For example, the self-assessment team reviewed Plant Review Board Minutes 95-29, dated December 1, 1995, which reviewed Justification for Continued Operation 95-06-00. Licensee personnel had identified and reported a condition potentially outside the design basis, which could lead to the turbine driven auxiliary feedwater pump tripping on overspeeed (Reference: Unresolved Item 528/9521-02). On January 10, 1996, licensee personnel approved the justification for continued operation for this issue. The justification for continued operation was prepared to provide information to be used in an operability determination for associated Condition Report/Disposition Request 9-5-0200.

On February 20, 1996, the self-assessment team requested the operability determination for this condition report/disposition request and was informed that it had not been initiated. As a result, Operability Determination 97 was prepared and Condition Report/Deficiency Request 9-6-0191 was written to evaluate and address why an operability determination was not performed when the justification for continued operation was written.

The team also reviewed a memorandum from nuclear regulatory affairs to operations, which documented several other operability determination issues related to the implementation of the operability determination program and establishing the operability determination basis. This included examples where an operability determination was not issued, and its basis was not established in a timely manner. Most of the examples noted in the memorandum were concerns originally identified by the NRC. The self-assessment team concluded that these issues, combined with the technical issues identified on two of the operability determinations reviewed from the auxiliary feedwater system, indicated that a larger problem existed with the recognition and performance of operability determinations. The self-assessment team concluded that operability determinations were occasionally treated as after-thoughts instead of first-order-of-business actions.

The licensee prepared Condition Report/Deficiency Request 9-6-0300 to evaluate and address interface issues between the following programs: the operability determination program, the condition report/disposition request program, the justification for continued operation program and the 10 CFR 50.59 program. Licensee personnel planned to clarify the applicable procedures. They planned to provide management expectations to operations personnel concerning the scope of operability determinations. They also provided training for site personnel which emphasized that it is necessary to initiate a corrective action document for degraded and nonconforming conditions to ensure followup and closure. In their media campaign licensee personnel stressed the importance of reporting degraded conditions to the control room.

Procedure 40DP-90P26, "Operability Determinations," requires that an operability decision be made within 24 hours of when a non-conforming condition is identified. The failure to complete the operability determination associated with Justification for Continued Operation 95-06-00 within 24 hours is a violation of Technical Specification 6.8.1. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

The self-assessment team also noted that specific problems identified on condition report/disposition requests were generally corrected, but repetitive and/or related problems were not always thoroughly analyzed to determine if more extensive evaluation or corrective action was needed.

4 INDEPENDENT NRC INSPECTION (40501)

4.1 Inspection Scope

The inspectors reviewed the licensee's self-assessment report, the detailed audit checklists, the information in the self-assessment team request for information notebooks, and the associated condition report/disposition requests to develop an understanding of the basis for the self-assessment team's conclusions.

The inspectors also reviewed portions of the design basis manual for the diesel generator and auxiliary feedwater system and applicable portions of Updated Final Safety Analysis Report.

The inspectors toured portions of the auxiliary feedwater system and the emergency diesel generator system with the cognizant system engineer and the cognizant self-assessment team member. The inspectors reviewed the self-assessment team's system walkdown deficiency reports. In addition, the inspectors performed an independent tour of portions of the auxiliary feedwater system and the condensate transfer system.

The inspectors interviewed self-assessment team members and other cognizant licensee personnel. The inspectors also attended self-assessment team meetings and the self-assessment team exit.

4.2 Observations and Findings

The inspectors generally agreed with the conclusions of the self-assessment team. The inspection activities, which resulted in a divergent or amplifying view are described below.

4.2.1 Auxiliary Feedwater Mini-flow Line Insulation

During the independent inspection of the auxiliary feedwater system and the condensate transfer system, the inspectors noted that the exposed portions of the safety related mini-flow return lines for the essential auxiliary feedwater pump were not insulated like the similar mini-flow return line for the nonessential auxiliary feed water pump. The inspectors requested the licensee to provide the basis for this difference.

The licensee initially determined that the installed configuration of the essential mini-flow lines (i.e., not insulated) was not consistent with the general guidance for freeze protection provided in Arizona Nuclear Power Project Mechanical General Design Criteria, Part II, Section 6.10, Revision 13.

On May 7, 1996, the inspectors telephoned licensee personnel to discuss the results of the licensee's investigation of the significance of this finding. Licensee personnel had performed additional analysis and determined that the installed configuration of the essential mini-flow lines was acceptable. The design criteria included an exception, which allowed insulation not to be installed if partial blockage due to freezing was acceptable. Licensee personnel calculated that for the design basis freeze (24 hours at 25° F), the expected partial freezing would not prevent the mini-flow lines from performing their protective function. Licensee personnel stated that they believed the architect engineer had intentionally omitted the insulation from these lines, although they had no particular basis for this belief.

The inspectors discussed with the licensee the fact that the inspectors found a potential hardware deficiency, which was not identified by the self-assessment team. The licensee determined that this oversight was caused by a system boundary change. The mini-flow return piping for the essential auxiliary feedwater pumps was designated as being a part of the condensate transfer system. The licensee stated that the self-assessment team stopped their tour when they reached the auxiliary feedwater system boundary.

4.2.2 Use of Probabilistic Risk Assessment Information

The inspectors reviewed a condition report/disposition request, which related to repeated tripping of the nonessential auxiliary feedwater pump due to suction pressure switch problems. Despite the fact that the affected equipment was risk significant, the licensee had not classified this condition report/disposition request so that a root-cause analysis would be performed. The inspectors discussed condition report/disposition request classification with the licensee and found that the licensee had downgraded the condition report/disposition request classification so that a root-cause analysis was not required because there was no specific Updated Final Safety Analysis

Report. Chapter 15 safety function for the pressure switch or the pump. While consideration of the risk significance was noted in the condition report/disposition request, the inspectors found that risk implications were not conservatively factored into the licensee's classification of the condition report/disposition request.

The inspectors noted that despite the repetitive nature of the pump trips, personnel from instrument and controls engineering had not been included in the team assigned responsibility for resolving the problem. The inspectors considered the downgraded condition report/disposition request classification to have contributed to the lack of involvement by instrument and controls engineering personnel.

4.2.3 License Basis for Nonessential Auxiliary Feedwater

The auxiliary feedwater system consisted of three trains of equipment for providing cooling to the steam generators in the event of a loss of main feedwater. Although originally designed as non-safety related, the nonessential train was modified during licensing to augment its reliability as a defense-in-depth design feature for accident mitigation. The Technical Specification Limiting Conditions for Operation were the same for the nonessential and the essential auxiliary feedwater pumps. The nonessential pump capability (with mini-flow secured) was described in the basis section of the Technical Specifications as equivalent to the flow required for the essential auxiliary feedwater pumps (650 gpm to a steam generator at 1270 psia). The nonessential train of auxiliary feedwater was also described in the Updated Final Safety Analysis Report; however, it was not specifically credited in any Chapter 15 analysis for accident mitigation.

The inspector noted that licensee personnel had not specified design basis flow requirements for the nonessential train of auxiliary feedwater for accident mitigation. The licensee's design basis manual for the auxiliary feedwater system stated that there is no safety analysis or design basis requirement that the non-essential auxiliary feedwater pump actually deliver 650 gpm to a steam generator at 1270 psia. The individual plant evaluation stated that the non-essential train of auxiliary feedwater is capable of 650 gpm, which is consistent with the Technical Specifications. The licensee stated that only 350 gpm was needed to meet the individual plant evaluation analysis criteria; they also stated that only 500 gpm was need to meet the analysis associated with the functional recovery procedures analysis.

Further, in NRC Inspection Report 50-528/95-21; 50-529/95-21; 50-530/95-21, the NRC identified that the licensee did not consider the capability to promptly open the discharge valves for the nonessential train of auxiliary feedwater following a main steam isolation signal actuation to be a design basis safety function of the valves. On November 27, 1995, following a main steam isolation signal actuation, it took operators 4 hours to open these valves. The inspector concluded that the licensee's design basis requirements did not ensure timely availability of the nonessential train of auxiliary feedwater system for accident mitigation.

The inspector noted that the licensee's design basis requirements for the nonessential train of auxiliary feedwater were not consistent with the risk significance of the equipment. The inspector noted that the nonessential train of auxiliary feedwater ranked high in significance within the licensee's probabilistic risk analysis. The licensee's individual plant evaluation stated that the single highest dominant contributor to the unavailability of auxiliary feedwater system is the human failure to restore the nonessential train of auxiliary feedwater following a main steam isolation signal actuation.

The inspector concluded that the licensee considered only the Updated Final Safety Analysis Report Chapter 15 analysis to define design basis requirements and safety functions. The inspectors concluded that the licensee had not consistently translated the licensing basis for the nonessential train of the auxiliary feedwater system from the basis section of the Technical Specifications into the design basis for the train. The licensee verbally committed to clarify their position with respect to the use of the nonessential train of auxiliary feedwater. This concern will be an inspection followup item (50-528/9601-01; 50-529/9601-01; 50-530/9601-01).

4.2.4 Design Basis Validation Not Comprehensive

The inspectors noted that the condensate transfer system, which included the condensate storage tank and the auxiliary feedwater mini-flow lines, was needed to accomplish the safety functions specified for the auxiliary feedwater system. Both the self-assessment team and the inspectors identified minor design discrepancies related to the condensate transfer system. During followup discussions with licensee personnel, the inspectors learned that while the licensee had developed a design basis manual and performed a design basis validation for the auxiliary feedwater system, they had not performed a similar review for the condensate transfer system.

To address this weakness, licensee personnel stated that they planned to complete a design basis manual for the condensate transfer system and perform a design basis validation of approximately 20 percent of the manual.

4.2.5 Lack of Formal Prompt Operability Determinations

The inspectors identified two additional examples of operability determinations, which were not performed for potentially nonconforming items. The self-assessment team originally identified both technical issues, but did not follow through to ensure prompt operability determinations were performed, because they believed the equipment to be operable. In both cases, engineering personnel were actively resolving the technical issues and believed there was a technical basis to support operability. Following discussions with the inspector, the licensee performed a prompt operability determination for both technical issues and found the equipment to be operable.

The inspectors viewed these instances as additional examples of an overall weaknesses in the prioritization of operability determinations, which was identified by the licensee's self-assessment team in Section 3.2.9.

5 UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) IMPLEMENTATION

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for additional verification that licensees were complying with UFSAR commitments. During an approximate 2-month time period all reactor inspections will provide additional attention to UFSAR commitments and their incorporation into plant practices, parameters and procedures.

While performing the inspections which are discussed in this report the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The self-assessment team identified several minor inconsistencies between the wording of the UFSAR and the plant practices, procedures and/or parameters. They identified the deficiencies in their corrective action system. The inspectors did not identify any additional examples of UFSAR discrepancies.

ATTACHMENT 1

PERSONS CONTACTED AND EXIT MEETING

1 PERSONS CONTACTED

1.1 Arizona Public Service Company

J. Bailey, Vice President, Nuclear Engineering
B. Endsor, Visitor, Nuclear Electric
F. Gowers, Site Representative, El Paso Electric
R. Henry, Site Representative, Salt River Project
J. Hesser, Director, Nuclear Engineering
M. Hodge, Section Leader, Nuclear Engineering
D. Kanitz, Senior Engineer, Nuclear Regulatory Affairs
A. Krainik, Department Leader, Nuclear Regulatory Affairs
D. Leech, Section Leader, Nuclear Assurance Engineering
M. Powell, Department Leader, Nuclear Engineering
C. Seaman, Director, Nuclear Assurance
G. Shanker, Department Leader, Nuclear Assurance Engineering

1.2 NRC Personnel

K. Brockman, Deputy Division Director, Division of Reactor Safety
J. Kramer, Resident Inspector, Division of Reactor Projects
C. Myers, Reactor Inspector, Division of Reactor Safety
L. Smith, Reactor Inspector, Division of Reactor Safety

The personnel listed above attended the exit meeting. In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on April 26, 1996. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee did not express a position on the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors. On May 7, 1996 the NRC further discussed the insulation requirements for the mini-flow lines associated with the essential auxiliary feedwater trains. On May 24, 1996 the NRC reviewed the overall conclusions of the inspection report with licensee management. Licensee personnel agreed to provide a commitment in writing to clarifiy their position on the use of the nonessential train of auxiliary feedwater.

JOHN D. STAMM

EDUCATION & TRAINING:

- B.S., Mechanical Engineering, Kansas State University, 1976

PROFESSIONAL REGISTRATIONS AND CERTIFICATIONS:

- Professional Engineer, Missouri, E-19644

EXPERIENCE:

4/81 to Present — **Wolf Creek Nuclear Operating Corp.**
Wolf Creek Generating Station

Summary
Multiple positions held at Wolf Creek Generating Station covering a broad range of Engineering duties and responsibilities beginning during the plant construction phase, continuing through startup, power ascension, and power operations. During my tenure at WCGS, I have held the following positions.

Supervisor, Safety Analysis
Responsible for supervision of the Safety Analysis and Probability Safety Assessment groups. USAR Chapter 15 accident analysis, thermal hydraulic analysis and risk assessment techniques are performed in support of in-house core design and other plant activities.

Division Manager, System Engineering
Responsible for administering the NSSS, BOP, Auxiliary, and Electrical systems groups whose job functions assured system health including plant trending, prioritization of system activities, generation of plant modifications, operability determinations, reportability evaluations, and screening/assignment of field generated documents.

Division Manager, Engineering Support
Responsible for administering the Project Engineering, Configuration Management, ASME, Design/Drafting, and Design Bases groups.

Manager, Plant Design Engineering
Responsible for administering the onsite Mechanical, Electrical, and Stress/Civil Engineering groups. Functions included development of design changes, performance of operability determinations and reportability evaluations in support of plant operations, and screening/assignment of all plant generated documents to the Engineering Department. Additionally, the administration of A/E support for major projects was performed.

Manager, Project Engineering
Responsibilities included supervision of the Project Engineering, Estimating, and Scheduling groups as well as the supporting clerical staff who developed the annual capital budget; developed the scope, schedule, and cost estimates for all proposed projects valued over $25K; prioritized and assigned all work documents to the Engineering department, and developed cost/benefit analysis for proposed plant modifications.

JOHN D. STAMM

Lead Mechanical Design Engineer
Accountable for review/approval of design changes and supervision of the site mechanical design group.

Lead Shift Test Engineer
Supervised the power ascension test crew throughout Initial Core Load, Low Power Physics testing, and Power Ascension testing required for commercial operation.

Senior Engineer
Performed construction inspection activities, coordinated Initial Surveillance test procedure write-up for the IST, HVAC, ILRT/LLRT activities; developed the initial plant performance monitoring program; and wrote, reviewed, and performed pre-operational test procedures.

10/77 - 2/81 **Performance Testing & Consultants, Inc.**

Vice President and 25% Shareholder
Administered the following projects:

- Monthly Heat Rate testing of 10 separate Electric Generating Stations for a midwestern utility.
- Air pollution compliance, efficiency, and acceptance testing of pollution control equipment for various electric generating stations, hospitals, and industrial facilities throughout the country.
- Energy and Technical Assistance Audits performed under the National Energy Audit Policy Act of 1978, Title III, sponsored by the Department of Energy.

Also served as Personnel Manager and participated in managerial duties such as business development, computer programming, estimating, proposal and technical report writing.

8/72 - 10/77 **Burns & McDonnell Engineering Co.**

Mechanical Engineer
Air Quality Control Division. Participated in design of a flue gas desulfurization system for a 170 MW unit in Illinois and the FGD system and electrostatic precipitators for three 600 MW units in Wyoming. Served as Test Director for EPA compliance tests at seven generating stations in Kansas, Missouri, and Kentucky.

Cooperative Education Student
Alternated semesters while working towards my engineering degree. Participated in source testing, ambient air testing, computer based dispersion modeling, and technical writing of Environmental Impact Studies.