APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-313/94-11
50-368/94-11

Licenses: DPR-51
NPF-6

Licensee: Entergy Operations, Inc.
Route 3, Box 137G
Russellville, Arkansas

Facility Name: Arkansas Nuclear One, Units 1 and 2

Inspection At: Russellville, Arkansas

Inspection Conducted: February 7-11, 17-18, and March 17, 1994

Inspector: P.A. Goldberg, Reactor Inspector, Engineering Branch, Division of Reactor Safety

Approved: T. F. Westerman 3-30-94
T.F. Westerman, Chief, Engineering Branch, Division of Reactor Safety — Date

Inspection Summary

Areas Inspected (Unit 1): Routine, announced followup inspection of licensee's actions in response to previously identified issues identified in NRC Inspection Report 50-313/89-200.

Areas Inspected (Unit 2): Routine, announced inspection of licensee's actions in response to deviations from code conformance found during walkdowns of the isometric update program.

Results (Unit 1 & 2):

- The licensee's actions in response to the previously identified items in NRC Inspection Report 50-313/89-200 were found to be good (Section 2).
- The licensee's engineering documentation was found to be thorough and provided adequate reference to design basis requirements (Section 2).

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- The licensee is pursuing review of the seismic qualification under operational loads of the Unit 1 feed water motor operated isolation valves with the vendor. The licensee is also performing the same review for other motor operated valves in Units 1 and 2 (Section 2.14).

- As a result of the licensee's walkdowns for the isometric update program, a number of code deviations are being identified. The schedule for final resolution is the fall of 1996 for Unit 1 and the spring of 1997 for Unit 2 (Section 2.19).

Summary of Inspection Findings:

- The following inspection followup items were closed:

  313/89200-01
  313/89200-02
  313/89200-03
  313/89200-04
  313/89200-05
  313/89200-06
  313/89200-07
  313/89200-08
  313/89200-09
  313/89200-10
  313/89200-11
  313/89200-12
  313/89200-13
  313/89200-14
  313/89200-15
  313/89200-16
  313/89200-17
  313/89200-18

- The following inspection followup items were opened:

  313/9411-01; 368/9411-01 (Section 2.14)

  313/9411-02; 368/9411-02 (Section 2.19)

Attachment:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 INTRODUCTION

An inspection of the licensee's actions in response to items identified during the inspection conducted to verify compliance with NRC Bulletin 79-02 (anchor bolts and baseplates) and 79-14 (piping analysis consistency with plant configuration) was conducted from February 7 through 11, 1994. The Unit 1 NRC Bulletins 79-02 and 79-14 inspection was documented in NRC Inspection Report 50-313/89-200. The inspector used the guidance of NRC Inspection Procedure 92701 for followup items.

2 NRC INSPECTION REPORT 50-313/89-200 FOLLOWUP INSPECTION

2.1 (Closed) Inspection Followup Item 50-313/89200-01: Sample Size of Anchor Bolts to be Tested

Bulletin 79-02 required that bolt samples be selected randomly and tested to achieve a 95 percent confidence level, that less than 5 percent defective bolts were installed in any one of the safety-related piping systems, and the sampling program conducted on a system-by-system basis. However, ANO had randomly selected and torque tested bolts from the whole plant, not on a system basis, without increasing the size of the sample. As a result, the total number of bolts sampled should have increased by a factor of 10 in order to comply with the bulletin.

As an alternate approach, instead of additional torque testing, the licensee augmented the original bolt testing program. The alternate approach incorporated an anchor bolt inspection during walkdowns on safety system piping in conjunction with the Isometric Update Project (IUP). The walkdown of the anchor bolts included verification of the anchor bolt size, type, thread engagement, snug tightness checked by hand, and angle of installation. One hundred percent of the accessible anchor bolts will be walked down during the IUP program. At the time of this inspection, the IUP project was approximately 70 percent complete. Records indicated that only 0.08 percent of the bolts inspected failed to meet the IUP criteria. In addition, ANO is approximately 10 percent complete with the walkdowns associated with "Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment," dated February 14, 1992. During the performance of this procedure, the anchor bolt tightness of safety-related equipment was verified by torquing the anchor bolts with a wrench.

The inspector concluded that between the IUP walkdowns and the program for torquing the anchor bolts with a wrench, the licensee was adequately addressing the inspection requirements of NRC Bulletin 79-02.

2.2 (Closed) Inspection Followup Item 50-313/89200-02: Piping Analysis Nonconservatisms for Pump Nozzle Loads

NRC Inspection Report 89-2'0 had identified three modeling discrepancies in the decay heat removal suction piping analysis, which, if corrected, would have increased the nozzle loads on the decay heat removal pump and the reactor building spray pump. In addition, the effect of eccentricity of mass for an actuator was not considered in the analysis model. Valves were modeled as equivalent pipe with twice the nominal wall thickness, and flanges were modeled as pipes with additional mass.

The licensee revised Calculations 88-E-0140-27, Revision 2, "Nozzle Review for Decay Heat Pumps P34A and P34B Suction and Discharge," and 87-D-1098-02, Revision 5, "Piping Analysis for Decay Heat Removal Pump Suction Lines," using a more realistic temperature gradient. The results of the reanalysis showed that the decay heat removal suction piping and pump nozzles met their design requirements.

For the eccentricity of mass of the manual valve which had a large handwheel gearbox, the licensee prepared Calculation 90-E-0020-01, Revision 0, "Study of Manual Valve BW-8A, 8B, DH1A and 1B Operator Eccentricity." The calculation concluded that the eccentricity had negligible effect with an increase of 1 percent on the nozzle loads. For the generic concerns of eccentricity of manual valves, the licensee prepared a Position Paper SPP-088-0, Revision 0, "Modeling of Mass Eccentricity of Large Manual Valves," which defined the approach for assuring consideration of manual valve mass eccentricity for future piping analysis. This position paper applied to new calculations and revisions to existing calculations.

The licensee prepared Position Paper SPP-299-0, Revision 0, "Thermal Expansion Considerations in Modeling of Valves and Flanges." This position paper reviewed the modeling approach used for valves and flanges and determined the impact on the thermal analysis. The position paper concluded that the modeling practice was appropriate, and the impact on thermal expansion was negligible.

The inspector concluded that the calculation revisions had shown that the original analysis methods had been acceptable. In addition, the position papers provided guidance for more conservative calculations in the future.

2.3 (Closed) Inspection Followup Item 50-313/89200-03: Seismic Anchor Motion

In NRC Inspection Report 50-313/89-200, the inspectors noted that seismic anchor motion was not considered in the piping analysis of the emergency feedwater turbine steam supply line attachment points to the main steam piping. The inspectors also noted that seismic anchor motion was not considered on the safety-related main feedwater piping from the containment penetration through the containment isolation valve. In addition, the inspectors were concerned that ANO had not evaluated other safety-related piping analyses to ensure that piping models which terminated at non-rigid piping and anchors had been correctly modeled to include seismic anchor motion.

In response, the licensee prepared Calculation 87-0-1099-10, Revision 0, "Main Steam Pipe to the EFW Turbine Pump and the Atmospheric Dump System Requalification." This analysis considered seismic anchor motion and the results indicated that the stresses were less than Code allowables. The licensee prepared Calculation 89-E-0086-01, Revision 0, "Operability Assessment for Main Feedwater Line Outside Containment," and 89-E-0086-02, Revision 0, "Operability Call for Main Feedwater Pipe Supports." These calculations addressed the exclusion of seismic anchor motion and found the piping to be within Code allowables with all of the supports operable.

ANO performed a study per EAR 89-0358 to determine if seismic anchor motion was consistently considered in ANO calculations. The initial review was conducted on 72 calculations and then expanded to a 100 percent review. Six analyses were identified as not having considered seismic anchor motion greater than the 1/8-inch, cut-off value used in the survey. Condition reports were generated and were evaluated with the seismic anchor movement load cases included. All piping and supports met code allowables with no modifications.

The licensee prepared Position Paper SPP-306-0, Revision 0, "Seismic Anchor Motion," which provided justification for the practices followed during Unit 1 design relative to consideration of seismic anchor motion. The paper stated that the applicable code of record did not specifically address or require consideration of the effects of seismic anchor motion on piping stress. From 1970 to 1978 the effects of seismic anchor motion on piping was not documented in the analysis which was what the licensee felt was general industry practice at that time. ANO currently considers seismic anchor motion if it was considered in the previous analysis or if the motions are greater than 1/16 of an inch for analysis done in conjunction with upgrades or enhancements to plant systems.

The inspector considered that the licensee had adequately addressed this followup item.

2.4 <u>(Closed) Inspection Followup Item 50-313/89200-04: Thermal Expansion for Varying Operating Modes</u>

In NRC Inspection Report 50-313/89-200, it was noted that ANO had not identified all associated emergency feedwater turbine steam supply operating modes which might require thermal expansion analysis. In addition, the report stated that ANO had not provided assurance that other safety-related piping analyses included thermal expansion load cases.

ANO prepared Calculation 87-0-1099-10, Revision 0, "Main Steam Pipe to the EFW Turbine Pump and Atmospheric Dump System Requalification," which included additional thermal modes. The calculation concluded that the piping system was in code compliance with the addition of the thermal modes.

Position Paper SPP-301-0, Revision 0, "Operating Modes for Thermal Expansion," was prepared. The paper documented the practices of performing thermal analyses during the Unit 1 design phase. The paper stated that a uniform maximum temperature had been used in analysis which the licensee felt was consistent with industry practices at that time. In addition, during startup walkdowns were performed during hot functional tests. The systems did not exhibit any thermal expansion-related problems.

The licensee stated that they were in process of developing pressure and temperature calculations which would identify various operating modes to be considered in future analysis. The licensee stated that work is in progress and may be completed by the end of 1994.

The inspector concluded that the licensee had adequately addressed this followup item.

2.5 (Closed) Inspection Followup Item 50-313/89200-05: Zero Period Acceleration (ZPA)

NRC Inspection Report 50-313/89-200 stated that the inspectors had reviewed four analyses, and only two had properly considered ZPA. The licensee stated that the practices used by ANO concerning consideration of ZPA in the original design of Unit 1 was consistent with the regulatory requirements and PSAR commitments at that time. ANO stated that for reconciliation analyses it was not a requirement to consider ZPA. ZPA has been used on a case-by-case basis since the early 1980s at ANO. Currently Design Specification APL-M-2514, Revision 2, "Technical Specification for the Design of Piping," requires that ZPA be considered for new design work.

The inspector concluded that the licensee had adequately addressed the followup item.

2.6 (Closed) Inspection Followup Item 50-313/89200-06: Eccentric Mass of Valve Actuators

NRC Inspection Report 50-313/89-200 identified two examples where Unit 1 piping analyses had failed to consider the eccentric mass of valve actuators. The report was also concerned about the generic implications of not considering the eccentric mass of valve actuators on other safety-related systems. One of the examples identified is discussed in Section 2.2 of this report. The second example was the main feedwater containment isolation valve. The piping was reanalyzed (Analysis 87-0-1-99-10, dated August 6, 1990) considering the valve eccentricity and the pipe stress levels increased by less than one percent. The licensee concluded that the discrepancy had an insignificant impact on the piping qualification.

ANO reviewed 107 seismic calculations to determine if there was a generic concern regarding the modeling of the eccentric mass of the valve actuators. The review showed that eccentricities of valve actuators were considered in piping analyses, with the exception of the two examples identified in the NRC report. Based on the results of the review, the licensee concluded that the two calculations were isolated cases. In addition, ANO Revised Specification M-2514, Revision 2, to include a section on eccentric masses which stated that valves with extended operators must be addressed by either actual modeling or documentation of the negligible effects of such eccentricities on the analysis.

The inspector concluded that the licensee had adequately addressed the followup item.

2.7 <u>(Closed) Inspection Followup Item 50-313/89200-07: Containment Penetration Displacement</u>

NRC Inspection Report 50-313/89-200 identified that containment penetration displacements had not been considered with post LOCA temperature rise and pressurization of the reactor containment building during a review of the decay heat removal suction piping analysis. ANO stated that not considering containment anchor movements is consistent with their understanding of other plants the same age as Unit 1, especially with a similar containment design. Specification SES-15, Revision 0, "Engineering Standard," stated that the containment penetration movements due to post-LOCA temperature and pressure rise were less than 1/8 inch and produced stresses of a secondary nature. The licensee evaluation as documented in Specification SRS-15, Revision 0, was that movement up to 0.15 inches was bounding. The licensee stated that the integrated leak rate test motions were also less than 1/8 inch.

The inspector concluded that the licensee had adequately responded to this followup item.

2.8 <u>(Closed) Inspection Followup Item 50-313/89200-08: Nonfunctional Pipe Supports in the Service Water System</u>

NRC Inspection Report 50-313/89-200 identified two pipe supports in the service water system which were not load bearing due to a 1/4-inch gap between the stanchion and the floor. ANO personnel reexamined the supports and determined that only one of the supports did not provide enough bearing to act as a vertical restraint. The support was corrected per Job Order 790167, dated July 14, 1989. Calculation 89-E-0083-12, Revision 0, "Operability Assessment for Intake Structural Service Water," was prepared and determined that the system was operable without the support.

The NRC report requested that ANO should include in the reanalysis consideration of all sliding supports that did not use a friction reducing material. The licensee reanalyzed the section of service water pipe. The licensee stated that since the deflections in the unrestrained directions were less than 1/16 inch friction was not a consideration. The licensee stated that friction forces were considered in the support design for all reanalysis or new designs. However, for reconciliation type analyses, friction loads on supports were not considered unless it was considered in the original analysis.

ANO had discovered during an operability review that the wrong response spectra curves had been used by the architect/engineer during design of the service water system. The licensee performed an operability evaluation which determined the system was operable and installed Design Change Package DCP 89-1036 during refueling outage 1R9. The installation of the design change package resolved all code compliance issues with the system.

The inspector concluded that the licensee had adequately addressed this followup issue.

2.9 (Closed) Inspection Followup Item 50-313/89200-09: Recently Reworked DHR Pipe Support Not in Agreement with Design

A walkdown of the decay heat removal suction piping had identified that a wide flange box restraint had a 1/4-inch gap at the top of the pipe instead of the design value of 1/16 inch. The licensee reviewed the piping analysis for the decay heat suction pipe where the support was installed and found that the pipe support did not experience load in the positive upward direction. The licensee concluded that the pipe support was acceptable from an operability and code qualification standpoint. In addition, the licensee noted that QC had identified the excessive gap on the support and initiated Condition Report CR-1-89-0069, dated February 4, 1989, prior to the NRC inspection.

The inspector concluded the licensee had adequately responded to the followup item.

2.10 (Closed) Inspection Followup Item 50-314/89200-10: Spring Hangers

During walkdowns, numerous discrepancies had been identified with regard to spring hanger design. The inspectors had determined that the licensee did not have an evaluation of all of the discrepancies including timeliness of maintenance activities. The inspectors also concluded that the licensee did not have a program to ensure that spring settings were verified and maintained within acceptable tolerance.

Plant Procedure 1092.023, "ASME Section XI Visual Examinations," was revised to provide appropriate tolerances for spring settings. Inservice Inspection Procedure 5120.241 was revised to require evaluation for all relevant indications including out of tolerance spring cans. Spring cans not inspected by the inservice inspection program were inspected under the maintenance program. The maintenance procedures were also revised to include tolerances. In addition to providing tolerances, the procedures addressed the timeliness concerns for potentially significant discrepancies.

The inspector concluded that the licensee had revised their program to include an evaluation of discrepancies and had a program to ensure that spring settings were verified and maintained within an acceptable level.

2.11 (Closed) Inspection Followup Item 50-313/89200-11: Snubber Settings

A concern was expressed in NRC Inspection Report 50-313/89-200 about the timeliness of the implementation of the ANO snubber reconciliation program. The purpose of the snubber reconciliation program was to update the pipe support drawings in accordance with the existing stress analysis and create a data base to augment the inservice inspection evaluation of snubbers. Implementation for the Unit 1 snubber reconciliation program was completed and documented in ANO Memorandum ANO-91-00623, dated February 25, 1991. All Unit 1 snubbers had been reviewed. The licensee had determined that for Unit 2 an extensive program was not necessary since the Unit 2 snubber drawings were in better condition than the Unit 1 drawings.

The licensee issued Structural Engineering Standard SES-22, Revision 0, "Snubber Design Criteria," dated January 5, 1993, which contained guidelines for utilizing snubbers in piping systems. The guidelines included snubber selection, design and documentation requirements for snubber supports.

The inspector concluded that the licensee had adequately addressed the concerns expressed in the NRC Inspection Report.

2.12 (Closed) Inspection Followup Item 50-313/89200-12: Main Feedwater Containment Isolation Valve Interaction with Structural Platform

During a walkdown, the NRC inspection team had noted that the actuator of a main feedwater isolation valve was in contact with the handrail of the ladder of a structural platform. A concern was also expressed about what programs were in place for Unit 1 to review unacceptable seismic interactions. The licensee initiated Job Order JO 00791941, dated January 26, 1990, which corrected the interference.

The licensee revised Specification ANO-M-2410, Revision 5, "Installation, Modification, Inspection and Documentation of Piping Systems and Pipe Supports, Hangers and Restraints," to include a section which required that a space clearance envelope of 12 inches was maintained around piping components. The licensee stated that this requirement is applicable for new analyses and new designs. For existing systems, the isometric update program used clearances from other standards.

The inspector concluded that the licensee had adequately addressed the followup item.

2.13 <u>(Closed) Inspection Followup Item 50-313/89200-13: Main Feedwater Water Hammer Analysis</u>

In NRC Inspection Report 50-313/89-200, ANO was requested to provide assurance that the safety-related portion of the main feedwater piping would maintain its pressure integrity subsequent to a water hammer resulting from a design basis seismic event. In addition, the report stated that without an analysis or justification, there would be no assurance that the emergency feedwater system could meet its licensing design requirements. Another concern was expressed in that other systems might not have considered water hammer loading.

An evaluation and walkdown of the ANO, Unit 1, turbine building was performed by a consulting engineering firm. The purpose was to assess the seismic ruggedness of the building at the maximum earthquake level, safe shutdown earthquake. The consulting firm reviewed a number of calculations and drawings of the turbine building, as well as performing a walkdown. The consulting firm concluded that the turbine bui'ing had sufficient ruggedness to withstand the maximum earthquake with virt ly no damage. This was documented in Report Number 93C1793, dated Ja ry 28, 1994, from Stevenson and Associates. The report also stated that .. was not a credible scenario that a seismic event would cause a catastrophic failure of the turbine building which could cause a guillotine failure of the main feedwater piping located in the turbine building.

ANO prepared Report 91-R-1016-03, Revision 0, "Postulated Failures Due to Design Basis Seismic Event," which assessed transient events which could occur as a result of a seismic event. A main feedwater line break on both legs or a loop was postulated. Although neither event, as a result of a seismic event, was a part of the licensing basis, both were assessed. The report determined that the addition to core damage frequency was calculated to be 3E-8. Based on this, the report concluded that the scenario was insignificant, and no further action was required. In addition to the analysis, the report stated that the emergency feedwater piping is seismically qualified and had no common portion with the main feedwater system. Therefore, the emergency feedwater systems ability to deliver flow to the steam generators was not expected to be affected by a seismic event.

Bechtel Power Corporation performed a review of similar piping arrangements for Unit 1 and identified emergency feedwater and service water of potential concern for water hammer loading. Since emergency feedwater was found to be a moderate energy line, a pipe crack would be postulated, not a line break. Water hammer in the service water piping is being extensively evaluated and is being tracked under another inspection followup item.

The Office of Nuclear Reactor Regulation staff brought to the attention of the license personnel and the NRC inspector during a March 17, 1994, telephone call, that water hammer had been classified by the NRC to be an unresolved safety issue (USI) and classified as USI A-1, "Water Hammer." Subsequently, USI A-1 was considered resolved by publication of NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," in March 1984. Subsequent staff reassessments have not changed the resolved status of USI A-1. The inspector concluded that based on the NRC staff's position on USI A-1, further inspection of main feedwater water hammer is not required. The item is closed.

2.14 (Open) Inspection Followup Item 50-313/89200-14: Seismic Qualification of Main Feedwater Isolation Valve

NRC Inspection Report 50-313/89-200 identified that the main feedwater isolation valves were not qualified for the orientation that they were installed. The valve actuators had been installed in the horizontal position and qualified by the vendor in the vertical position. The licensee prepared Calculations 89-E-0086-01 and -02, Revision 0, which addressed the operability of the main feedwater piping with horizontal actuators. The results of the calculations indicated that the valve orientation had an insignificant impact on the qualification of the piping.

Calculations V-CV-2630-05, Revision 3, "Seismic Qualification of Valve Assembly CV-2630, and V-CV-2680-05,"; and Revision 1, "Seismic Qualification of Valve Assembly CV-2680," were performed to evaluate the valves for seismic and operational loadings simultaneously. The results of the calculations determined that the valves were not rigid and were qualified for a thrust slightly greater than the required thrusts for the specific valves. However, the qualified acceleration was stated to be substantially below the as-built piping accelerations. The licensee stated that they would treat these valves as deviations from code compliance, pending review of the seismic qualification of these valves by the valve vendor.

The licensee stated that they had approximately 80 to 85 motor operated valves with discrepancies due to increases in thrust values. In addition, they stated that there may be a few additional valves that may not be able to open or close during a seismic event. The licensee agreed to review their licensing basis to determine the requirements for the motor operated valves. This issue was identified as Inspection Followup Item 313/9411-01; 368/9411-01.

2.15 (Closed) Inspection Followup [illegible] 50-313/89200-15: Damaged Decay Heat Removal Piping

NRC Inspection Report 50-313/89-200 reviewed a decay heat removal system water hammer event where deficiencies such as pipe dents and failed lugs had occurred. The report requested that ANO review the local stresses on all lugs attached to the decay heat removal piping and that ANO confirm that local strain hardening did not occur in the area of the dented pipe. ANO committed to perform hardness testing.

The licensee prepared Condition Report CR-1-89-0069, dated February 27, 1990, which caused a review of lug attachments to be performed. The review identified a total of 16 pipe supports with lug attachments. The licensee determined that 15 of the supports met code requirements. The other support was determined to be operable, but was modified during refueling outage 1R9 in accordance with Design Change Package DCP 89-1029. Calculation 87-D-1098-21, Revision 1, "Evaluation of Local Denting on 6 Inch Schedule 10S DHR Piping at Support GHB-2-DH-209," was prepared to evaluate the decay heat removal dented piping. The calculation recommended that the dents in the piping could be left as is. This recommendation was made since the weak link in the piping was the welds, not the dents.

The inspector concluded that the licensee had adequately addressed the issue.

2.16 (Closed) Inspection Followup Item 50-313/89200-16: Code Reconciliation

The NRC inspection had identified that ANO had been performing reanalyses of piping systems in accordance with a later edition of the ASME Code Section III without performing any reconciliation between the later code and the code of record. The licensee prepared SPP-025-0, Revision 0, "Arkansas Nuclear One Units 1 and 2 Piping Design Code Reconciliation," dated September 28, 1990, for code year reconciliation. Specification M-2514, Revision 1, was also issued in September 1990, which recommended piping analysis codes for new designs.

Based on these documents, the inspector concluded that the licensee had documents for reconciling code years.

2.17 (Closed) Inspection Followup Item 50-313/89200-17: Updating Stress Analysis Calculations

The NRC inspection team found that their review of the stress analysis for the emergency feedwater turbine steam supply was hindered by the fragmented status of the calculation. A number of calculations had to be considered jointly to assess the qualification of the piping. In response to this concern, the licensee consolidated the series of emergency feedwater calculations into Calculation 87-D-1099-10, Revision 0, dated August 6, 1990. The licensee also stated that several other calculations with similar problems have either been consolidated, or will be in the near future.

Based on the revised calculations, the inspector concluded that the licensee had adequately responded to the followup item.

2.18 (Closed) Inspection Followup Item 50-313/89200-18: Minor Discrepancies

NRC Inspection Report 50-313/89-200 identified a number of minor discrepancies which were identified by the inspection team. These discrepancies were examples of differences between the as-built and as-designed piping that had not been reconciled or documented prior to the inspection. These discrepancies were found in the main feedwater system, emergency feedwater system, emergency feedwater turbine steam supply system, decay heat removal system, and service water system. Many of the discrepancies dealt with differences between drawing dimensions and installed dimensions, spring hanger scales not reading within tolerance, weld size differences and clearance dimensions.

The inspector reviewed all of the discrepancies listed in the 89-200 report and concluded that the licensee had satisfactorily resolved all of the issues.

2.19 (Open) Inspection Followup Item 50-313/9411-02; 50-368/9411-02: Deviations from Code Conformance

During the followup inspection, the inspector found that the licensee had identified a number of deviations from code conformance while performing walkdowns for the isometric update program. A number of these deviations were found during walkdowns during one outage and then deferred to either the next outage or a later outage. The inspector questioned the length of time between the discovery of the deviations and their resolution.

The licensee stated that most of the deviations were pipe support related and fell into two categories: repair issues and design discrepancies. Repair issues, which included items such as incomplete welds and missing members, were processed in accordance with the licensee's job order system. The licensee stated that each issue was reviewed and, if it were a safety concern, it would be completed in the outage it was discovered. Otherwise, the deviation would be deferred to the next outage. Design issues, which included reanalysis and modifications, were processed in accordance with the licensee's EARs. The licensee stated that an operability evaluation was performed for each EAR and a Category 1, 2, or 3 was assigned. A Category 1 was assigned if engineering judgement was the only method to determine operability. A Category 2 operability assessment was based on an analytical approach, but also used engineering judgement. A Category 3 operability assessment was based on an analytical approach. The licensee stated that Category 1 items were completed by the next refueling outage. Category 2 and 3 items were often deferred one outage or more. The basis for the deferrals was time constraints for outage work and the desire to identify all code deviations for a stress problem and complete them all at the same time.

The licensee stated that there are currently 77 open issues for Unit 1 and 77 open issues for Unit 2. In addition, the licensee stated that completion of the Unit 1 deviations is scheduled for the fall of 1996 and completion of the Unit 2 deviations is scheduled for the spring of 1997. The licensee stated during the telephone discussion on February 17, 1994, that they plan to submit a letter to the NRC to formalize their schedule for completion of the code deviations. The completion of the code deviations will be reviewed during a future inspection and has been identified as Inspection Followup Item 313/9411-02; 368/9411-02.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee personnel

* S. Bennett, Acting Supervisor, Licensing
* W. Greeson, Supervisor of Structural Analysis
* R. Lane, Director, Design Engineering
* J. Martin, Senior Engineer
* D. Mims, Director, Licensing
* P. Novero, Supervisor, Stress Analysis
* S. Pyle, Licensing Specialist
* W. Rogers, Supervisor, Design Engineering
* D. Saunders, Project Manager, Isometric Upgrade Program
* C. Turk, Manager, Mechanical, Civil, Structural

1.2 NRC Personnel

* J. Melfi, Resident Inspector

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

* Denotes personnel that attended the exit meeting.

2 EXIT MEETING

An exit meeting was conducted on February 11, 1994. During this meeting, the inspector reviewed the scope and findings of the inspection. 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 inspector.