**ENCLOSURE 2**

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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| Docket Nos.: | 50-313; 50-368 |
| License Nos.: | DPR-51; NPF-6 |
| Report No.: | 50-313/97-21; 50-368/97-21 |
| Licensee: | Entergy Operations, Inc. |
| Facility: | Arkansas Nuclear One, Units 1 and 2 |
| Location: | Junction of Hwy. 64W and Hwy.333 South<br>Russellville, Arkansas |
| Dates: | October 27-31 and November 10-14, 1997 |
| Inspectors: | M. Runyan, Team Leader<br>P. Goldberg, Reactor Inspector<br>R. Bywater, Reactor Inspector<br>G. Kalman, Project Manager |
| Accompanying Personnel | D. Denver, Consultant |
| Approved By: | T. Stetka, Acting Chief, Engineering Branch, Division of Reactor Safety |
| Attachment 1: | Supplemental Information |
| Attachment 2: | List of Documents Reviewed for Unresolved Item 50-313/97201-06 |

## EXECUTIVE SUMMARY

Arkansas Nuclear One, Units 1 and 2
NRC Inspection Report 50-313/97-21; 50-368/97-21

### Engineering

- Within a review of selected documents, the licensee's 10 CFR 50.59 safety evaluation program appeared to be functioning satisfactorily, with the exception that some documented word searches of the Updated Safety Analysis Report for impact were inaccurate. However, some apparent failures to implement the program were identified (Sections E2.1 and E8.22).

- The failure to procedurally incorporate a vendor recommended maximum emergency feedwater flow limit of 1500 gpm to a single steam generator into plant operating procedures was identified as a violation of 10 CFR Part 50, Appendix B, Criterion V (Section E8.7).

- The failure to properly verify the piping configuration design inputs to a design calculation was identified as the first example of a violation of 10 CFR Part 50, Appendix B, Criterion III (Section E8.8).

- Overpressure of the turbine-driven emergency feedwater pump downstream piping required further reviews by the licensee. The existing analysis used pump performance criteria that was less than the manufacturer's specified criteria (Section E8.9).

- The failure to properly verify the design adequacy in four design calculations was identified as additional examples of a violation of 10 CFR Part 50, Appendix B, Criterion III (Section E8.11).

- The lack of a review by the licensee of a sample of existing design calculations to determine if the types of past problems identified by the licensee existed within other calculations was identified as a weakness (Section E8.11).

- The lack of over current trip testing of Unit 1 molded case circuit breakers required further review by the NRC (Section E8.16).

- Although the NRC identified five discrepancies in the Final Safety Analysis Report (FSAR), the licensee's ongoing review of the FSAR was considered sufficient to correct these specific examples and to resolve the resulting generic concern. Therefore, enforcement discretion was taken in accordance with the NRC Enforcement Policy (Section E8.18).

- The failure to provide measures to ensure the correct translation of seismic design requirements to instrument tubing installations and the maintenance of this configuration during plant operations in Unit 1 was identified as an example of a violation of 10 CFR Part 50, Appendix B, Criterion III (Section E8.20).

- Three apparent violations were identified associated with activities involving the removal of the borated water storage tank vacuum relief valve for maintenance and testing on December 4, 1996. The apparent violations involved the failure to perform proper safety evaluations as described in 10 CFR 50.59, the failure to perform the vacuum relief valve testing under the plant conditions as prescribed in the test procedure in accordance with 10 CFR Part 50, Appendix B, Criterion V, and the failure to utilize the correct design control processes (temporary alterations) for the temporary cover configurations placed over the valve flange in accordance with 10 CFR Part 50, Appendix B, Criterion III (Section E8.22).

- The failure to properly check the adequacy of an engineering report that failed to account for instrument error in a calculation of vortexing in the borated water storage tank was identified as an example of a violation of 10 CFR Part 50, Appendix B, Criterion III (Section E8.23).

Plant Support

- During a plant walkdown, the NRC identified a fire barrier deficiency and the licensee took prompt compensatory actions as required (Section F2).

## Report Details

Summary of Plant Status

Units 1 and 2 were operated at 100 percent power during the inspection.

## III. Engineering

### E2 Engineering Support of Facilities and Equipment (37001)

E2.1 10 CFR 50.59 Implementation

a. Inspection Scope (37001)

The inspectors reviewed the licensee's program to change plant design, revise procedures, and conduct tests and experiments without prior NRC approval as described under 10 CFR 50.59. The licensee's program was reviewed for compliance with regulatory requirements and implementation of the program was evaluated by reviewing three 10 CFR 50.59 evaluations. The inspectors reviewed the following documents:

- ANO-1 10 CFR 50.59 Summary Report for 1996, dated May 22, 1997
- ANO-2 10 CFR 50.59 Summary Report for 1996, dated June 19, 1997

50.59 Evaluations:

- ANO-1, Procedure Change 95-7023, "Reactor Coolant Pump Starting Temperature"
- ANO-1, Condition Report 1-96-0567, "Reactor Coolant Pump Lift Pump Disabled"
- ANO-2, Design Change Package 94-2007, "Replace Charging Pump Downstream Check Valves CVC-22A, B, and C"

Procedure 1000.131, "10 CFR 50.59 Review Program," Revision 3

b. Observations and Findings

Within the review of the listed documents, the inspectors did not identify any programmatic deficiencies in the licensee's 10 CFR 50.59 program. It was noted that the licensee chose to evaluate the "margin of safety" of proposed changes by evaluating the changes only against the basis section of the Technical Specifications. This practice had been previously questioned by the NRC in Inspection Report 50-313;368/97-04 and

found to be in compliance with 10 CFR 50.59 (however, a violation for a failure to perform a safety evaluation was issued). No deficiencies in the "margin of safety" evaluations were identified by the inspectors during review of the 10 CFR 50.59 evaluations. Although currently in compliance, the licensee indicated that a more encompassing design basis for margin of safety determinations will be implemented in the future.

While review of these safety evaluations indicated appropriate programmatic controls, some apparent failures to implement the 10 CFR 50.59 program were identified as discussed in Section E8.22 of this report.

The inspectors also identified problems with the licensee's search capability for the Updated Final Safety Analysis Report. For example, Safety Evaluation 96-141 stated, "The lube oil pumps are not described in the Technical Specifications, Safety Analysis Report, or operating license." The inspectors determined that this was in error. The lube oil pumps were discussed in Section 4.2.2.6 of the Updated Final Safety Analysis Report. The inspectors identified similar problems with the safety evaluation determinations for the procedure changes. The inspectors discussed these findings with the licensee and concluded that the automated search methodologies used to search for key words in licensing basis documents were not always effective in finding relevant reference material and that the technical reviewers were not ensuring that the scope of review was adequate. The licensee agreed with these conclusions and informed the inspectors that they had also identified this issue in the past year and had revised the safety evaluation procedure to address the deficiencies. The inspectors review of Procedure 1000.131 indicated that improvements had been made in the licensee's process to ensure that adequate automated searches were performed and that they were reviewed by a second individual.

c. Conclusions

Within the review of selected documents, the licensee's 10 CFR 50.59 safety evaluation program, when implemented, appeared to be functioning satisfactorily, with the exception of Updated Final Safety Analysis Report key word searches. However, some apparent failures to implement the program were identified as discussed in Section E8.22 of this report.

**E8 Miscellaneous Engineering Issues (92903)**

E8.1 (Open) Unresolved Item 50-313/9623-01: Consideration of Multiple Hot Short Actuations

Background

This issue addressed the unplanned energization of motor-operated valves (MOVs) during a control room fire. This event, caused by a control cable shorting to an energized conductor, is known as a hot short. Under certain configurations, a hot-shorted MOV could stroke with the torque or limit switch removed from the control circuit, causing the valve to stall into its open or closed seat. Depending on the capacity of the

motor actuator and the strength of the valve, this action could damage the valve sufficiently such that plant operators would not be able to reposition the valve using the valve's manual handwheel. The licensee's emergency operating procedures addressing a control room fire and subsequent evacuation of the control room assumed that manual control of accessible valves was available. The licensee interpreted existing regulations to require postulation of only one hot short during a control room fire, whereas the NRC maintained that multiple hot shorts, which is defined as more than one MOV having a hot short at the same time, were within the scope of the regulations. Independent from the resolution of this matter, at the request of the NRC, the licensee agreed to perform a calculation to determine which valves would be potentially susceptible to hot short damage. This was the only aspect of this issue reviewed during the current inspection.

Inspector Followup

The inspectors reviewed Engineering Report 97-R-0004-01, "Engineering Information Notice 92-18 Evaluation," Revision 1 PC-1; Calculation 96-E-0065-01, "Unit 1 Safety Related Valve Survivability Evaluation Under Stall Thrust/Torque, Motor Operated Valves," Revision 1; and Calculation 96-E-0066-01, "Unit 2 Safety Related Valve Survivability Evaluation Under Stall Thrust/Torque, Motor Operated Valves," Revision 0.

The inspectors reviewed Calculation 96-E-0065-01. The purpose of this calculation was to determine operability with reduced margin for a select group of safe shutdown MOVs under stall thrust/torque conditions. The evaluations were intended to reflect one-time operation under the stated conditions and were not intended for use as design input in other analyses or calculations. An important assumption stated that the analysis was valid only for the time it was performed since actual test data was used in some evaluations. The results showed that 37 of the 66 valves evaluated would be able to be manually positioned to their safe position following postulated hot short conditions. Twenty-nine valves were susceptible to damage. The inspectors determined that the licensee had used a valid methodology for determining the maximum torque and thrust that an MOV would experience during a hot short event, with one exception. For MOVs that did not have previous measurements of stem friction, the licensee used a plant average stem factor correlated to a stem friction coefficient of 0.1342. The use of a bounding stem friction coefficient for the untested valve population, approximately 0.08, would have resulted in calculated torques and thrusts as much as 50 percent higher than those reported in the calculation. However, based on a qualitative judgement by the inspectors, application of the lower stem factors would not have significantly changed the number of valves determined as potentially incapable of being manually repositioned.

Engineering Report 97-R-0004-01 addressed multiple hot shorts, but in situations where two or more valves were determined to be unable to be manually repositioned, the report stated that it was not credible that both or all would fail at the same time in the same direction. The inspectors noted that this reasoning reiterated the licensee's position that multiple hot shorts were not required within the regulations.

Based on the results of the calculations and engineering report listed above and in light of the 29 valves in Unit 1 analyzed to be susceptible to hot shorts, the inspectors

concluded that some safety functions could be lost if multiple hot shorts occurred in a particular manner. This was due to the fact that some of the motor-operated valves may be subjected to loads during a hot short that would render them incapable of being manually stroked to their safe shutdown position. In some cases, as in the Unit 1 high pressure injection system, a total of eight valves would have to hot short in the same direction to defeat the safety function of the system. However, the Unit 1 shutdown cooling function could be lost with the hot-short failure of only two valves, CV-1428 and CV-1429, decay heat cooler outlet valves. Both valves were calculated to be potentially incapable of being manually repositioned following a hot short event.

The above discussion was illustrative of the potential impact of multiple hot short events and was not a complete listing of all potentially affected systems. This item will remain open pending further reviews by the NRC program office to define the regulatory requirements for this matter.

E8.2 (Closed) Violation 50-313/96027-03: Failure to Notify the NRC Within 1 Hour of Declaration of Notification of Unusual Event

Background

On October 17, 1996, the licensee declared a Notification of Unusual Event in accordance with its Emergency Plan for a fire in the Unit 1 reactor building that lasted more than 10 minutes. The licensee notified the NRC Operations Center of a fire in the reactor building within 1 hour of the declaration of the Notification of Unusual Event. However, the licensee did not inform the Operations Center that it had made a declaration of one of the Emergency Classes (Notice of Unusual Event) specified in its Emergency Plan in accordance with 10 CFR 50.72(a)(1)(i).

Inspector Followup

The licensee determined that the cause of the violation was human error. The shift engineer who made the initial contact with the NRC Operations Center provided details of the fire but neglected to inform the NRC that a Notification of Unusual Event had been declared.

The licensee conducted training with notification communicators about this event and NRC notification requirements. Additionally, the licensee reviewed applicable procedures and revised Procedure 1903.011, "Emergency Response/Notifications," Revision 22, to require verbatim transmission of information included on Form 1903.011Y, "Initial Notification Message," Revision 22, to the NRC Operations Center officer followed by transmitting the form to the NRC Operations Center via facsimile. The form explicitly identified the emergency classification for the event. The inspectors verified the licensee's corrective actions had been completed and considered these actions adequate to prevent recurrence of this violation.

E8.3 (Closed) Licensee Event Report 50-313/96-009: Fire in the Reactor Building During Heatup Resulted from a Cracked Weld in an Oil Line on a Reactor Coolant Pump Motor

Background

During a Unit 1 plant heatup on October 17, 1996, a fire occurred in insulation around the main feedwater nozzle ring on the Train B once through steam generator. A cracked weld in the discharge line of the Train B reactor coolant pump lube oil lift pump resulted in oil spray being introduced into the insulation. The hot piping caused the oil to ignite. A Notification of Unusual Event was declared, and fire brigade responders extinguished the fire approximately 16 minutes after discovery. The NRC conducted a special inspection to review the circumstances of the fire, continued safety of the plant, and corrective actions taken by the licensee. The results of the inspection are documented in NRC Inspection Report 50-313; -368/96-27. The licensee initiated Condition Report CR-1-96-0567, conducted an event evaluation, and documented its evaluation results in a root cause analysis report.

The licensee's event evaluation concluded that oil intrusion into the insulation caused a wicking effect that lowered the auto-ignition point of the oil to a lower than expected temperature.

The NRC conducted a special inspection documented by NRC Inspection Report 50-313; -368/96-27, which identified problems with the licensee's implementation of 10 CFR Part 50, Appendix R, fire protection requirements and that the licensee had failed to identify and correct problems that were noted prior to the fire. These were the subject of an NRC Escalated Enforcement Action (EA 96-512). The licensee developed and implemented many corrective actions to correct the deficiencies in these areas. They are discussed in Sections E8.4 and E8.5 of this report.

Inspector Followup

This item is closed based upon the followup and discussion in Sections E8.4 and E8.5 of this report, which address the escalated enforcement items associated with this event.

E8.4 (Closed) Enforcement Action 50-313; -368/96512-01013: Inadequate Lube Oil Collection Systems for Reactor Coolant Pumps

Background

During the NRC special inspection following the October 17, 1996, Unit 1 reactor building fire (NRC Inspection Report 50-313; -368/96-27), the NRC reviewed the lube oil collection systems for all of the reactor coolant pumps in both Units 1 and 2. The

inspectors determined that the lube oil collection systems for the Unit 1, Train B reactor coolant pump and all of the Unit 2 reactor coolant pumps had not been designed to collect oil leakage from all pressurized and unpressurized locations. Also, the Unit 1, Trains A, C, and D reactor coolant pumps had inadequate leakage collection from some unpressurized locations. The deficiencies were determined to be in violation of 10 CFR Part 50, Appendix R, Section III, requirements.

Immediately following the fire, the licensee placed the unit in cold shutdown to evaluate the effects of the fire and water suppression on plant equipment. No significant damage occurred other than to some insulation. The damaged insulation was repaired or replaced as necessary. Other actions taken prior to restarting the unit included installing a new shroud around the Train B reactor coolant pump motor and extending oil collection drip trays for all of the reactor coolant pump motors to provide coverage of potential leakage points. Due to remaining concerns regarding the integrity of other portions of the Unit 1, Train B motor lube oil piping, the licensee implemented administrative controls to prohibit operation of the associated high pressure lift oil pumps unless a fire watch was present. The motor manufacturer provided confirmation that the absence of lube oil injection following an inadvertent pump trip would not adversely affect the coastdown characteristics of the pump. This was used as supporting information in a 10 CFR 50.59 safety evaluation to conclude that operation without the lube oil lift pump in standby did not constitute an unreviewed safety question.

In Unit 2, the licensee performed lube oil collection system modifications following an unplanned reactor shutdown in November 1996. The NRC performed walkdowns of both units and reviewed the modifications. These activities were documented in NRC Inspection Reports 50-313; -368/96-27, 50-313; -368/96-07, and 50-313; -368/96-08. The NRC concluded that with the exception of the remote lube oil fill systems for both units, the lube oil collection system modifications established compliance with 10 CFR Part 50, Appendix R, requirements.

The Unit 1 remote oil fill lines (Tygon tubing) to the top of each motor were not provided with a leakage collection system as required by Appendix R requirements. The licensee provided administrative controls to prevent operation of the system until a modification and any necessary regulatory exemption was applied for and obtained. The Unit 2 remote oil system was designed differently, using stainless steel tubing and stainless steel flexible hose, but no oil collection system was provided. The licensee described the Unit 2 system in detail and proposed compensatory administrative controls for its use in an exemption request to the NRC dated December 23, 1996. The NRC approved the exemption request in a letter dated June 14, 1997. The exemption allowed the licensee to use the remote oil fill system without a collection system contingent upon the licensee's implementation of several compensatory measures prior to each use.

The licensee responded to the Notice of Violation in a letter to the NRC dated May 9, 1997. Corrective actions were identified in this document, as well as the licensee event report associated with this event, and several condition reports. Some of these were completed and reviewed previously as discussed above; others were reviewed during this inspection.

Inspector Followup

The inspectors reviewed documents, procedures, and interviewed personnel to determine the adequacy and completeness of the licensee's corrective actions. Also, the inspectors contacted the cognizant fire protection engineer in the NRC Office of Nuclear Reactor Regulation regarding the adequacy of the Unit 2 lube oil collection system. Documents reviewed included: portions of the Updated Final Safety Analysis Report; Condition Report 1-96-0567, its associated Root Cause Analysis Report and corrective action documents (including 10 CFR 50.59 safety evaluations and screenings); Procedure 1102.002, "Plant Startup," Revision 59; Procedure 1103.006, "Reactor Coolant Pump Operation," Revision 20; Procedure 1107.001, "Electrical System Operations," Revision 50; Procedure 1107.004, "Battery and 125V DC Distribution," Revision 8; and Procedure 1504.001, "Visual Inspection of the Unit 1 & 2 RCP's Oil Collection System," Revision 4.

Modifications were made to the lube oil collection systems for all reactor coolant pumps in both units. These were determined by the NRC to be acceptable for meeting 10 CFR Part 50, Appendix R, requirements. To ensure the continued operability of the systems, the licensee implemented Procedure 1504.001, for inspection of the systems immediately prior to refueling outages and prior to startup.

The licensee placed administrative controls on operation of the Unit 1, Train B reactor coolant pump lift oil pumps. This was done because of uncertainty about oil system integrity during lift pump operation on the Jeumont Industries motor. Procedure 1103.006 was revised to require that the Train B reactor coolant pump not be started or stopped during normal operations when the reactor coolant system temperature was greater than 350 degrees F. Operation of the lift oil system was allowed, but only if two fire brigade-trained operators were present in the reactor building to identify oil leakage, inform the control room, and extinguish any resulting fire.

The normal design of the oil lift system provided an autostart of the lift pump when the reactor coolant pump tripped. The administrative controls discussed above prevented an autostart from occurring. Therefore, if a reactor coolant pump trip occurred, the pump would coast down without injection of lubrication onto the motor thrust bearing. The inspectors reviewed the licensee's 10 CFR 50.59 Safety Evaluation 96-141, "Evaluation of Placing RCP #32B HP Lube Oil Pump (P-80B and P-63B) in Pull-to-Lock," and safety evaluation determinations for associated procedure changes. Supporting documentation for the licensee's evaluation included a letter from the vendor documenting that reactor coolant pump coastdown time would not be adversely affected for at least 15 seconds without oil lift pump operation. The relevant accident analyses for consideration in the safety evaluation were the various sizes of loss of coolant flow accidents, and the evaluation determined that the lack of lift pump operation had no impact. The inspectors concluded that the safety evaluation provided an acceptable basis for concluding that reactor coolant pump operation without availability of lube oil injection did not constitute an unreviewed safety question.

The licensee initiated an evaluation of the lube oil system piping configuration to ensure that no additional questionable welds remained. The licensee and the motor supplier determined that the lube oil piping system was constructed in accordance with applicable commercial standards with the use of qualified welders and adequate quality controls. No other similar weld failures were reported in more than 200 other reactor coolant pump motors manufactured by the motor supplier. One difference in the subject configuration was the addition of a backup direct current powered lube oil lift pump and associated piping. To evaluate this further, the licensee developed a plan to perform inservice visual inspections of the piping at operating pressure, collect vibration data, and perform data analysis. The licensee considered this option for addressing piping concerns preferable to others considered, which included piping replacement, weld replacement, and volumetric inspections with repair of rejectable welds. After discussing these options with the licensee, the inspectors considered the licensee's approach to evaluating the acceptability of the lube oil piping configuration to be adequate. Administrative controls on lift pump operation would remain in effect during future operating cycles pending completion of vibration analyses and continued visual inspections to identify any remaining weld failures.

The inspectors determined that the licensee had completed corrective actions to ensure that its reactor coolant pump oil collection systems were in compliance with 10 CFR Part 50, Appendix R requirements and that proposed corrective actions were either completed or in progress with responsive completion dates.

E8.5 (Closed) Enforcement Actions 50-313/96512-01023 and 50-313/96512-01033: Failure to Identify Significant Conditions Adverse to Quality and Take Prompt Corrective Action Relating to Oil Accumulation in Pipe Insulation.

Background

The NRC (and the licensee) determined that there were previous opportunities to identify that a problem existed with the reactor coolant pump oil lift system prior to the fire event. These opportunities were missed, in part, because a condition report was not initiated to thoroughly evaluate the extent of the potential problem. Some of the opportunities to identify the scope of the problem included: discovery of a crack in the lift oil pump discharge line, the magnitude of the crack and inadequacy of the original weld, discovery of oil on the side of the steam generator during reactor building walkdowns during plant heatup, discovery of oil puddled under the reactor coolant pump, and identification of an excessive amount of smoke present in the reactor building during heatup.

To correct these deficiencies in problem identification, communications, and understanding of oil-insulation interaction consequences, the licensee revised procedures and conducted training.

Inspector Followup

The inspectors reviewed documents, procedures, and interviewed personnel to determine the adequacy and completeness of the licensee's corrective actions.

Documents reviewed included: Procedure 1000.104, "Condition Reporting and Corrective Actions," Revision 13; Procedure 1015.036, "Containment Building Closeout," Revision 5; Procedure 2102.002, "Plant Heatup," Revision 10; Procedure 1102.002, "Plant Startup," Revision 61; Procedure 1102.010, "Plant Shutdown and Cooldown," Revision 41; Procedure 2102.010, "Plant Cooldown," Revision 30; Procedure 1203.034, "Smoke, Fire, or Explosion," Revision 10; Procedure 2203.034, "Fire or Explosion," Revision 5; and training outlines and attendance records for Condition Report 1-96-567.

The licensee's training appeared to adequately address the circumstances and causes of the fire and provided information on causes of insulation fires; uses and limitations of material safety data sheets; importance of proper delivery, receipt, and accuracy of communications; and lessons learned in response to oil spills and observations of smoke.

The condition reporting procedure was revised to clearly address identified deficiencies in fire protection system components including oil leaks and spills. Items of this type that meet the definitions in the procedure now require the initiation of a condition report. The other procedures provided requirements to perform inspections for oil leakage and excessive smoke and provided actions to take in response to identified adverse conditions. These actions could include terminating plant heatup, posting fire watch personnel with fire fighting equipment, contacting the fire protection engineer, and other appropriate actions. The procedures also contained guidance on the effects of oil soaked insulation on the auto-ignition temperature of the oil.

The inspectors concluded that the licensee's corrective actions were adequate to improve staff understanding of the consequences of oil leaks, and the importance of communications when adverse conditions are identified, provide actions to prevent fires from occurring, provide additional requirements for inspections, and provide actions to take when adverse conditions are identified.

E8.6 <u>(Closed) Inspection Followup Item 50-313/97201-01: Licensee's Actions to Revise the Technical Specification Bases for the Minimum Water Volume in the Condensate Storage Tank</u>

<u>Background</u>

The minimum water level in the Unit 1 safety-related condensate storage tank was based on the capability to provide a sufficient supply of water to permit the initiation of the decay heat removal system within 4 hours or less of an accident initiation. However, the actual period of time required to initiate decay heat removal may be substantially greater than 4 hours. The licensee acknowledged that this text was inaccurate and proposed to revise the existing Technical Specification bases to eliminate (probably) reference to the decay heat removal initiation and to state that the minimum condensate storage tank level was based on the volume that was required for 30 minutes of emergency feedwater operation before manual switchover of the pump suction to the service water system.

Inspector Followup

The licensee stated that they were in the process of developing the Improved Standard Technical Specifications and that the appropriate condensate storage tank bases would be included in the initial submittal. The licensee stated that they planned to submit the revised Technical Specification design base along with the revised Technical Specifications by the end of 1998. The inspectors verified that there was no safety concern related to the minimum condensate storage tank water level and that there was no need on the part of the licensee to revise this level. However, a clarification of the Technical Specification basis for the level was appropriate.

E8.7 (Closed) Unresolved Item 50-313/97201-02: Licensee's Failure to Incorporate Maximum Emergency Feedwater Flow Limits into Plant Procedures to Monitor and Preclude Exceeding Recommended Limits

Background

A 1991 Babcock and Wilcox Company engineering report (92-R-1019-01) identified that the maximum emergency feedwater flow rate to preclude damage to the Unit 1 steam generator tubes was 1500 gpm. The NRC determined that the licensee had not revised plant procedures to incorporate this upper flow limit. Specifically, Procedure 1010.002, "Transient History/Transient Cycle Logging," in effect at the time was not revised. The NRC reviewed operator logs and identified two occurrences where the emergency feedwater flow exceeded the 1500 gpm limit.

Inspector Followup

The inspectors reviewed Condition Report CR-1-97-0081, dated March 27, 1997, which the licensee generated to resolve the maximum emergency feedwater flow issue. A licensee contractor prepared an analysis which determined that the maximum feedwater flow rate to prevent steam generator tube damage was 2214 gpm. The inspectors reviewed the calculation and determined that the higher flow limit was adequately justified.

The licensee stated that the upper flow limit was not included in operating procedures because high emergency feedwater flows would cause a cooldown, during which operators would take procedural actions to correct the high flow rate problem. However, the inspectors determined that the licensee had not previously evaluated the effects of the high flow on the steam generator tubes for the periods of time before operator actions would restore an acceptable flow rate.

The inspectors reviewed Procedure 1010.010, "Unit One Transient Cycle Logging and Reporting," Revision 1, which was issued to provide guidance for determining excessive emergency feedwater flow and provide specifications for initiation of a condition report. This procedure superseded Procedure 1010.002. The inspectors determined that the Procedure 1010.010 required an engineering evaluation if the upper limit of the emergency feedwater flow was exceeded.

During the inspection, the inspectors reviewed in detail Framatome Technologies final draft calculation 32-1229968-00 and Framatome Technologies Calculation 32-5000450-00, "ANO Auxiliary Feedwater Capacity - Phase 2." The purpose of these calculations was to determine the limit for emergency feedwater flow to preclude unacceptable steam generator tube damage or wear. The results showed that a flow rate of 2214 gpm resulted in less turbulence induced vibration than at full power conditions. As part of this review, a conference call was held with Framatome Technologies and licensee personnel during which they responded to questions from the inspectors. Based on their review and the information presented during the conference call, the inspectors determined that the analysis provided in these calculations demonstrated that emergency feedwater operation at flow rates up to 2214 gpm for limited periods of time would have no significant effect on steam generator performance. The inspectors verified that design of the emergency feedwater system would not result in flow rates in excess of this amount.

The inspectors determined that the licensee failed to procedurally incorporate a vendor recommended maximum emergency feedwater flow limit of 1500 gpm to a single steam generator. As a result, flow rates in excess of this amount were permitted to occur for short periods of time. Later, in response to the NRC's questions, these excess flow rates were shown to be acceptable.

10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality be prescribed by and performed in accordance with instructions, procedures, or drawings. The licensee's failure to incorporate vendor recommended flow rate limits into procedures was identified as a violation of 10 CFR Part 50, Appendix B, Criterion V (50-313/9721-01).

Conclusions

The licensee failed to incorporate a vendor specified emergency feedwater flow limit into plant procedures, resulting in the failure to evaluate the effects of exceeding the vendor recommended flow limit during two previous periods of plant operation. Concerns related to possible damage to the steam generator tubes were adequately resolved by a subsequent analysis demonstrating the acceptability of higher flow limits.

E8.8 (Closed) Unresolved Item 50-313/97201-03: EFW Piping Configuration Differences

Background

In review of Calculation 82-D-2086-02, which evaluated net positive suction head requirements for the motor-driven and turbine-driven emergency feedwater pumps, the NRC identified that the suction piping configuration used in the calculation to determine pressure drops was not consistent with the piping configuration identified in the piping isometric drawings. The pressure drops in the piping were negligible and the effect on the results of the calculation due to the differences was small. This item was opened pending action to revise the calculation and take any appropriate generic actions.

Inspector Followup

The licensee initiated Action Item Requests L97-0198 and L97-0285 to revise Calculation 82-D-2086-02 to address the as-built configuration. The inspectors verified that these action items had been assigned to the appropriate engineering organization for completion. The licensee established a due date for completion of the calculation revision of March 1, 1998. Based on the minimal impact on the results of the calculation expected by incorporating the as-built piping configuration differences in the calculation, the inspectors considered the licensee's schedule for completion adequate. The broader issue of identified deficiencies in the licensee's maintenance of calculations is addressed in the inspectors' followup of Unresolved Item 50-313/97201-06 (Section E8.11).

10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that design control measures, such as design reviews, shall provide for verifying or checking the adequacy of design. The licensee's failure to properly verify that the design inputs for Calculation 82-D-2086-02 were consistent with the installed piping configuration was considered as the first example of a violation of 10 CFR Part 50, Criterion III (50-313/9721-02).

Conclusion

The licensee's failure to properly verify the design inputs to Calculation 82-D-2086-02 was identified as a violation of 10 CFR Part 50, Appendix B, Criterion III.

E8.9 (Open) Unresolved Item 50-313/97201-04: Inadequate Piping Pressure and Temperature Specifications

Background

During review of Calculation 88-E-0100-16, which evaluated pressure and temperature limitations of emergency feedwater system piping, the inspectors identified several discrepancies. For example, in the determination of maximum emergency feedwater discharge pressure limitations, the calculation used a pump suction alignment from the condensate storage tank instead of the service water system as input. The use of the service water system as a source would result in a higher emergency feedwater system discharge pressure than if suction were taken from the condensate storage tank.

The licensee initiated Engineering Request 973848 for the design engineering staff to review and revise Calculation 88-E-0100-16. The request was assigned a completion date of December 31, 1997.

Inspector Followup

The inspectors reviewed the licensee's followup action item response to the original inspection concern. The response identified that the design pressure of the emergency feedwater system discharge piping was 1850 psig. At rated speed of the turbine-driven emergency feedwater pump (3795 rpm) and with the suction aligned to the condensate storage tank, the calculated discharge pressure at shutoff head was 1732 psig. When the system was aligned to the service water system, the calculated discharge pressure at shutoff head was 1842 psig, resulting in a margin from the design pressure of only 8 psi. However, the calculation did not account for speed controller tolerances, nor did the calculation assume a failure of the speed controller. If a failure of the speed controller occurred, procedural guidance was provided for an operator to take manual control of the turbine-driven emergency feedwater pump. This procedure instructed the operator to not exceed a pump speed of 3900 rpm to avoid overpressurization of the discharge piping.

The licensee determined that at a pump speed of 3900 rpm, the resulting maximum discharge pressure was 1844 psig when suction was aligned to the condensate storage tank and was 1932 psig when suction was aligned to the service water system. Therefore, manual operation of the turbine-driven emergency feedwater pump at the maximum procedurally-allowed speed would result in a maximum discharge pressure that exceeded the design pressure of the piping.

The licensee's response stated that the utilization of the 1932 psig maximum pressure as a basis for discharge piping design pressure was not considered because manual pump control while aligned to the service water source would require an equipment failure in an already infrequent emergency plant condition.

The inspectors had a concern with this position. The condensate storage tank was designed to provide a tornado-protected 30-minute supply of water to the emergency feedwater system. The 30 minutes allowed the operations staff time to transfer the emergency feedwater supply to the service water system. However, if the turbine-driven speed control was out of tolerance or failed and the pump was operated manually per procedure, the maximum discharge pressure could exceed the design pressure of the turbine-driven pump discharge piping.

The licensee's response also addressed historical pump performance during testing at a pump speed of 3920 rpm. Analysis of the test results in Calculation 80-D-1083B-102A concluded that the discharge pressure would not have exceeded the 1850 psig design pressure, even if suction were aligned to the service water system at maximum pressure, because the pump was not operating at full potential. However, the pump had sufficient capacity to meet minimum specifications and, therefore, was operable.

The inspectors discussed these issues further with the licensee and were informed that the original calculation would be reviewed to determine the extent that higher-than nominal pump speeds need to be considered, both from a speed controller tolerance perspective and a manual operation perspective. Additionally, since the pump appeared to be under performing with respect to its certified pump curve, using the pump curve for

piping design pressure determination may be overly conservative. The inspectors were unable during this inspection to evaluate this issue further to determine whether the pump was performing in an acceptable manner with respect to the pump curve. However, the inspectors noted that the pump was meeting its Technical Specification limits. The licensee's due date for completion of this review and revision to the calculation was December 31, 1997. During the exit meeting conducted January 6, 1998, the licensee indicated that the due date for this effort had been extended to January 15, 1998.

The inspectors considered this item open pending further NRC review of the licensee's ongoing evaluation and review of the performance of the turbine-driven emergency feedwater pump.

E8.10 (Closed) Unresolved Item 50-313/97201-05: Evaluation of EFW Pump Room Environment

Background

The licensee identified an environmental qualification problem in the Unit 1 EFW pump room, which houses the turbine-driven and motor-driven EFW pumps. Steam traps located in the steam supply lines to the turbine-driven pump, which were not designed for seismic loads, were vented to the atmosphere of the room. A failure of one these traps (two in the high pressure piping and two in the low pressure piping) would cause a continuous blowdown of hot steam in the room, raising the ambient temperature and humidity, and possibly challenging the operability of environmentally-sensitive equipment in the room. The licensee took interim measures to leave one steam trap isolated and to permanently open a door to the room. The licensee then calculated the impact of a steam trap failure under this configuration assuming that operators could isolate the steam blowdown within 1 hour. The calculation (87-E-0026-09, Revision 0) concluded that temperature in the room would peak at 149.5 degrees F for a brief time, but that the steady state temperature would be 134 degrees F. Since the limiting long-term temperature for the room was 148 degrees F, the licensee concluded that the EFW system remained operable. The inspectors agreed with the licensee that humidity resulting from the event would not adversely affect equipment critical to the function of the EFW system.

This item was opened to track licensee efforts to make modifications to resolve this issue permanently and to make necessary revisions to related design basis documents. The licensee also indicated that an assessment of past operability of the EFW room would be performed.

Inspector Followup

In review of this finding, the licensee determined that the original design of the EFW steam traps was inadequate for both Units 1 and 2. The safety implications for Unit 2 were mitigated because the steam traps exhausted outside the EFW room. The licensee decided to modify EFW steam traps on both units: Unit 1 during Refueling Outage 1R14 (Spring 1998) and Unit 2 during Refueling Outage 2R13 (first quarter 1999).

The licensee performed a past operability assessment and concluded that the EFW system was operable in the past considering the room door closed, all traps in operation, and a 1-hour operator response to isolate the failed trap.

The inspectors reviewed Root Cause Analysis Report CR C-97-0048, "Unqualified Steam Trap Lines in the EFW Turbines' Steam Supply Lines." In this document, the licensee determined that the architect/engineer (A/E) for the design of the plant assumed that failure of the steam trap lines would not reduce the functional capability of the EFW systems. However, the A/E had evidently focused on sufficiency of the resulting supply of steam to the turbine and not on the environmental consequences of the failure.

The inspectors questioned the assumption that operator action to isolate steam would occur within 1 hour of a steam trap failure, an assumption essential to the past operability status of the system. The licensee stated that an operator surveillance of the room (checking lube oil flows) occurs by procedure soon after each automatic start of the EFW system, which would presumably result in identification of the steam trap failure. However, since no time parameter was specified for this surveillance, it could be indefinitely delayed if other complications were to occur. The inspectors asked about the effect of a response if the failure occurred after this surveillance effort. In this case, since no remote indication of temperature in the room is available, it did not seem reasonable to assume that steam isolation within one hour would occur. The licensee stated that the fire detection system would probably alarm given the amount of steam that would exist in the room. On one previous occasion, in 1992 (CR 1-92-0361), the deluge valve tripped open due to an excess amount of steam from the steam trap drain lines. However, on further discussion with the fire control engineer, the inspectors learned that the fire detection device in the EFW room is not sensitive to steam and would only possibly alarm in the event of a condensation buildup on the device. The inspectors determined that the fire detectors were not a reliable indication of a steam leak.

The evaluation of past operability indicated that, with the room door closed, the room temperature would reach 164 degrees F 1 hour after the steam trap failed. It seemed questionable whether an operator could enter an environment this harsh to perform the required valve isolation. In response to the inspectors' concern, the licensee generated a new action item (Item Number 9) for CR C-97-0048 to address the issue of accessibility as it pertained to the past operability determination. The licensee recalculated the room temperature profile using more realistic assumptions, including condensing heat transfer, and found that the room condition at 1 hour would be 133 degrees F at a relative humidity of 57 percent. This corresponded to a wet bulb globe temperature (WBGT) of 121.5 degrees F. At this WBGT, the licensee, based on US Navy

Document OPNAVINST 5100.19C, Change 1, dated May 15, 1996, determined that strenuous work at a maximum duration of 30 minutes could occur in this environment. The amount of time needed to isolate a failed steam trap would be only several minutes. The inspectors noted that the shape of the temperature profile suggested that the temperature would peak out at something less than 140 degrees F at a relative humidity of less than 60 percent. This suggested that the room would be fit for the necessary operator's actions to take place and that equipment operability would not be threatened.

During the inspection, the inspectors reviewed Calculation 87-E-0026-09, "EFW Pump Room Temperature Profiles." This calculation provided an analysis of the heat up of the EFW pump room assuming various failed-open steam traps in addition to the normal thermal load provided by the EFW pumps and other equipment. The analysis was carried out with an ANO developed and maintained software package, "RHUR," which calculates room heat up from arbitrary sources. RHUR was programmed in a "macro" language executed in a spreadsheet program, SYMPHONY. It was documented as a software package and was controlled and maintained under the software control program. The RHUR model developed for this analysis was constructed by substantially modifying the "macro" statements that make up the RHUR software package.

The inspector reviewed the analysis provided in the calculation and determined that the results of the four cases presented seemed reasonable and conservative for the given configurations. In response to the NRC concern, the licensee provided additional supporting analysis in the form of an alternate calculation using the GOTHIC software program that substantially verified the results presented in Calculation 87-E-0026-09.

The inspectors considered that the licensee had adequately established the past operability of the EFW room equipment. Based on the less severe conditions of having the room door removed, the inspectors also considered this information to support the licensee's assessment of current operability. However, the inspectors identified a weakness in the licensee's original response to this issue in that it failed to document consideration of the ability of an operator to perform an assumed operation in a harsh environment. A second weakness was the lack of verification of the software model used to calculate the postulated EFW room thermal transient. Although considered good engineering practice, licensee procedures did not specifically address this validation.

The licensee was in the process of developing a modification for both units to mitigate the effects of pressure boundary failures and the blowdown of live steam from steam traps in the EFW rooms, while preserving adequate condensate removal. Through discussions, the inspectors learned that the modification would reroute steam trap piping to exhaust outside the EFW room and upgrade the piping to be safety-related and seismically qualified. The inspectors determined that the modification would resolve the hardware concerns associated with this issue.

E8.11 (Closed) Unresolved Item 50-313/97201-06: Design Control Weakness

Background

This issue involved several calculation deficiencies. The examples included calculations containing plant configuration discrepancies as well as calculations that required enhancement, updating, or to be superseded. The specific examples were as follows:

- Calculation 88-E-0086-01, "IE Bulletin 88-04 Review for P7A and P7B Minimum Flow Evaluation," did not adequately document the effect of two parallel EFW pumps operating on minimum recirculation.

- Calculation 92-E-0077-04, "Unit 1 EFW System Pump Performance Requirements," used an incorrect level in tank T-41B for EFW actuation due to a loss of offsite power following a tornado.

- Calculation 92-E-0021-01, "Emergency Duty Cycle and Battery Sizing Calculation," was not revised to reflect plant modifications that resulted in changes in the continuous and locked rotor current for EFW Valves CV-2620 and CV-2627, and a modification that resulted in an additional load due to the reactor building spray pumps close and trip circuits.

- Calculation 80-D-1083A-02, "EFIC DC Valve Torque Calculation Under Reduced Voltage Condition," was not superseded as required.

The actions implemented or planned by the licensee included those to correct the specific examples and to improve design control programs to prevent recurrence.

Inspector Followup

The inspectors reviewed Licensee Information Requests (LIRs) L97-0289 and 0290 as well as Condition Reports CR-C-97-0058 and CR-C-97-0059.

LIR L97-289 addressed the failure to update Battery Loading Calculation 92-E-0021-01, following implementation of Design Change Packages (DCPs) 92-1003 and 93-008. The licensees's response to the inspection report (letter 1CAN099703, dated September 22, 1997), indicated that this calculation would be superseded before January 15, 1998. However, at the time of this inspection, this calculation had already been updated and was reviewed by the inspectors and found acceptable. In discussions regarding this inconsistency, the licensee explained that the oversight resulted from the existence of a large backlog of data that had not been entered into the Design Configuration Information Management System (DCIMS). This condition resulted in a failure to identify affected calculations when engineers accessed the DCIMS system to perform searches

based on key words related to a particular modification or calculation revision. The licensee indicated that this backlog had been eliminated and that current practice would preclude new backlog formation. The inspectors confirmed that no backlog existed and that the DCIMS was up-to-date at the time of this inspection. In addition, the inspectors reviewed Memorandum ANO-97-00225 issued to the engineering staff that reinforced the existing procedures and guidance related to calculation updates for load changes.

The second issue within LIR L97-289 dealt with Calculation 80-D-1083A-02, "EFIC DC Valve Torque Calculation Under Reduced Voltage Condition" which should have been identified as superseded. The licensees's response to the inspection report indicated that this calculation will be superseded before January 15, 1998. Discussions with licensee personnel and review of the applicable procedure, OP-5010-015, "Engineering Calculations," indicated that clear direction existed to ensure that superseded calculations are identified. The cause of the specific instance was attributed to a backlog of data to be entered in the primary tracking tool, DCIMS, that had previously existed. As discussed above, the inspectors confirmed that no backlog existed at this time of this inspection.

LIR L97-290 addressed Calculations 88-E-0086-01 and 92-E-0077-04, both of which needed enhancement for various reasons. The licensees's response to the inspection report indicated that these calculations would be revised before March 1, 1998.

10 CFR Part 50, Criterion III, "Design Control," states that measures shall be established to ensure that applicable requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

The licensee's failure to properly account for design information in the Calculations 88-E-0086-01, 92-E-0077-04, 92-E-0021-01, and 80-D-1083A-02 was considered as additional examples of a violation of 10 CFR Part 50, Appendix B, Criterion III (50-313/9721-02).

The inspectors review of Condition Report CR-C-97-0058 indicated that the condition report dealt with prioritization and completion of the Upper Level Documents (ULD) and ULD-related discrepancies. The actions taken as a result of the condition report included determining completion priority and establishing and communicating to the engineering organization the philosophy for use of the ULDs. The inspector reviewed the actions taken on these issues and confirmed that they were appropriate.

The inspectors review of Condition Report CR-C-97-0059 indicated that this condition report dealt with completion of the Design Configuration Documentation (DCD) Project and resolution of discrepancies identified during this project. The project was completed in December 1994; however, approximately 100 discrepancies remained unresolved. The discrepancies were originally screened for significance and operability, but this CR recognized that recently there was greater interest in and higher standards being applied to design configuration issues. In addition, several other recent CRs have been initiated to reexamine previously closed discrepancies that were found to have inadequate resolution by current expectations. The CR-C-97-0059 evaluation established an action

to prioritize review of the remaining open discrepancies. This action was completed. In addition, previously closed discrepancies have been reviewed to ensure that they are adequate. This completed action led to further actions to evaluate 44 previously closed items. These subsequent actions were not available for review at the time of this inspection.

In addition to the specific observations and related corrective actions reviewed above, the inspectors reviewed other condition reports, procedures, guidelines, and programmatic evaluations that impact the control of calculations. These are listed as Attachment 2 to this report. Several recent calculations and design change packages were reviewed to assess the results of ongoing improvement efforts on current calculation practices. These actions addressed broader issues associated with calculation processes and control.

The licensee initially identified a concern regarding the adequacy of their calculation processes in Condition Report CR-C-96-0060 initiated in March 1996. The most significant outgrowth of efforts to resolve this concern led to the formation of a Quality Action Team (QAT) to study calculation process issues. The team's final report, published in September 1996, identified a large number of concerns (56) organized into the following broad categories:

- Calculation in-processing time was too long creating backlogs and delays for engineering personnel.

- The DCIMS data base contained inaccurate, incomplete, not up-to-date or erroneous information.

- The calculation preparation, revision, and approval process requirements were not clearly and consistently defined, documented, or understood by engineering personnel.

- Retrieval of existing calculations was sometimes difficult.

- Inconsistent interpretation of standards for legibility of calculation documents resulted in delays for processing calculations through quality assurance records storage.

- Data may be buried within a calculation with its relationship to other processes not defined.

- Program engineers were not always aware when other engineers were revising calculations that impact their program.

In reviewing the QAT action plan, the inspectors determined that the proposed corrective actions were appropriate and that they should provide significant calculation process improvements when fully implemented. In addition, the interim compensatory actions

that were implemented in the short term appeared to be of sufficient quality to reduce the occurrence of calculation deficiencies.

In addition to reviewing the licensee's action plan, the inspectors reviewed a sample of approximately 30 calculations generally chosen at random. These are listed in Attachment 2. The inspectors found that the calculations were thorough, generally clear, and well documented. Assumptions and other data sources were clearly identified and referenced. The few exceptions were found in some older calculations performed during original plant design and construction, approximately 20 years ago.

The inspectors focused on the licensee's process for assigning status to calculations. This included the use of new, pending, as-installed, amending, and superseded status levels. Although the inspectors found the use of the amending status somewhat confusing and cumbersome, no erroneous cross references were found in the sample reviewed. For example, there are no limits to the number of calculations that can amend a given calculation. Nor is there a time limit beyond which amending calculations would require incorporation. In practice, however, the only calculations found with a large number of amending calculations were the electrical loading calculations. In this case, the licensee's practice was to assign a coordinator or owner for these calculations. The coordinator was responsible for incorporating all amending calculations once each cycle. The incorporated calculations were then assigned a status of superseded. The inspectors reviewed and found this to be correctly performed for a recently issued revision to Calculation 92-E-0021-01, "Emergency Duty Cycle and Battery Sizing Calculations." In addition, as mentioned above, the licensee instituted a Calculation Revision Notice (CRN) process that allows minor changes to calculations to be issued and more easily tracked. This process was designed to reduce the number of amending calculations to no more than nine. In practice, based on a sample review, the inspectors found no examples with more than two CRNs.

In spite of an apparently comprehensive effort to improve the calculation process, the inspectors recognized a potential inconsistency in the way the licensee handled the issue. The licensee's quality assurance team identified a concern related to the preexistent backlog in the DCIMS data base that had apparently been the cause of several NRC-identified discrepancies. In addition, other persistent problems within the calculation process were identified by the team. However, the licensee did not perform a sample review of existing calculations to determine whether the DCIMS problem, in addition to the other identified problems listed above, had created a situation where a large number of calculations currently in effect may contain technical errors as a result of these problems. The inspectors considered the lack of review of a sample of calculations to represent a weakness in the licensee's overall response to the issue.

The inspectors reviewed the capability and use of the DCIMS database. It was a very useful and powerful tool for searching for needed information concerning calculations, evaluations, and reports.

Conclusions

A violation of 10 CFR Part 50, Appendix B, Criterion III, was identified concerning discrepancies identified with four licensee calculations. The lack of a review of existing calculations to determine if past problems identified by the licensee affected other calculations of record was identified as a weakness.

E8.12 (Closed) Inspection Followup Item 50-382/97201-07: Licensee's Actions for Revising the Modification Procedure to Address Field Routed Installations

Background

The NRC identified that the layout configuration of certain installed conduits was different from that shown on the conduit and cable tray layout drawings. The NRC also identified that there were no drawings or records of the seismic support details for the installed conduit and cable tray supports. The licensee stated that this was due to conduits being field routed.

The NRC noted that conduits attached to the wall of the emergency feedwater pump room contained redundant cables supported on the same unistrut support. The licensee stated that the supports were designed to withstand seismic design loads and were passive. In addition, the licensee stated that redundant trains could be supported from a common support from a seismic standpoint provided Appendix R and other criteria were addressed.

The licensee's modification procedure discussed a constructability walkdown involving the modification engineer, modification supervisor, craft supervisor, and quality engineer. During discussions, the licensee stated that they would document the decisions arrived at during the walkdown by including these in the form of work steps in the controlled work package. The licensee stated that this would include conduit routing and seismic support details that would be documented as sign-off steps. The need to revise the modification procedure to address field routed installations was identified as Inspection Followup Item 50-313/97201-07.

Inspector Followup

The inspectors reviewed the licensee's response to the inspection followup item documented in Letter 1CAN099703, dated September 22, 1997 (response letter to NRC Inspection Report 50-313/97-201). The inspectors also interviewed licensee personnel. The inspectors requested that the licensee supply details concerning how the Appendix R requirements were met for the redundant cables supported on the same unistrut in the emergency feedwater pump room.

The inspectors reviewed draft Calculation 85-E-0086-1, "Safe Shutdown Capability Assessment," Revision 3, and found that the turbine driven emergency feedwater pump cable and the motor driven emergency feedwater pump cable had a 1-hour fire barrier installed. In addition, the inspectors found that an NRC exemption was granted for

configurations having less than 20 feet of separation. The inspectors reviewed licensee Letters 1CAN068706, dated June 24, 1987, and 1CAN048708, dated April 22, 1987, which requested the exemption for the cables. The exemption, granted in an NRC letter dated October 26, 1988, contained a stipulation that a 1-hour fire wrap was necessary in addition to a fire suppression system. During a walkdown, the inspectors determined that there was a fire suppression system in the pump room and that the cable was wrapped with fire wrap.

The inspectors reviewed the revised Procedure 6030.005, "Control of Modification Work," Revision 3, Change PC-1. The inspectors noted that the procedure was revised to include instructions for the development of a single line drawing depicting the types of supports used, the support spacing, and the conduit layout. The procedure required that the drawing be included with the controlled work package.

E8.13 (Closed) Inspection Followup Item 50-313/97201-08: Licensee's Actions to Perform a Revised Cable Pulling Calculation Using Additional Design Information Obtained from the Cable Vendor

Background

The NRC identified that Procedure/Work Plan 6030.112, "Installation of Raceway Systems," used for the installation of raceways, included a table of conduit bend radii for manufacturers' standard bends but not for long radius bends. The NRC asked the licensee which cables required long radius bends as a result of the minimum bending radius and maximum sidewall pressure restrictions during pulling. The licensee provided a calculation that compared the cable minimum bending radii to the bend radii of the conduit that was appropriate for a single cable. The NRC questioned the values for the radii on the inside surface of the conduit bends as well as the accuracy of the cable diameters and maximum cable bend radii. The licensee stated that they would obtain updated cable vendor data to verify the conduit bend dimensions and would revise the calculation accordingly. The NRC reviewed the preliminary calculation and noted that a few large diameter cables would require long radius bends. This item was opened pending licensee actions to obtain additional design information from the cable vendor in order to perform a revised cable pulling calculation.

Inspector Followup

The inspectors reviewed the licensee's response to the inspection followup item in Letter 1CAN099703, dated September 22, 1997, and interviewed licensee personnel. The inspectors also requested a listing of condition reports involving damage to cables that resulted from cable pulling; however, no such conditions were found.

The licensee stated that updated cable information would be obtained from cable vendors for approximately 10 percent of the cable codes listed in the cable minimum bend radius review table. The licensee expected to receive this information by mid-January, 1998. The licensee also stated that the updated cable information obtained would be used to revise the calculation for each of the applicable cable codes. The licensee stated that the cable calculations would be revised by March 1998.

The inspectors determined that the licensee's corrective actions were adequate for the current closure of this item based on the fact that the condition report search had not found any cables damaged by cable pulling and that the actions necessary to revise the cable code calculations were being tracked and had acceptable completion dates.

E8.14 (Closed) Inspection Followup Item 50-313/97201-11: Verification of Calculation Changes Associated with DC Batteries

Background

The NRC identified that Calculation 92-E-0021-04, "Battery D06 and D07 Recharge Time," Revision 0, was not revised to show the reduced charging time resulting from a change out of the Unit 1 200 amp battery chargers with new 400 amp chargers. An amending calculation, 93-D-1010-04, "Recharge Time for D06 and D07 with Changes per DCP-93-1010," Revision 0, calculated the revised charging time. This item was opened pending incorporation of the amending calculation information into the parent calculation.

Inspector Followup

The inspectors determined that this condition did not represent an actual problem because the battery recharge times were not required by license conditions and did not affect battery capacity. In addition, as discussed in Section E8.11 of this report, there were no licensee procedures requiring incorporation of amending calculations into the parent calculation within a specific time period. Based on discussions with the licensee, the inspectors determined that information provided in amending calculations, but missing from the parent calculation, should not result in a quality concern because annotations provided in the parent calculation positively identified the existence of the amending calculations.

The inspectors verified that amending Calculation 93-D-1010-04 had been incorporated into Calculation 92-E-0021-04. The inspectors observed that the revised recharge times were based on the updated 400 amp recharge capacity.

E8.15 (Closed) Inspection Followup Item 50-313/97201-12: Review of Current Design Basis for Bypassing Emergency Diesel Generator (EDG) Protective Trip

Background

For emergency starts of the Unit 1 EDGs, the licensee did not bypass those protective functions delineated by Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby Onsite Electrical Power Systems at Nuclear Power Plants," to be bypassed under accident conditions. This was because the licensee was committed to Safety Guide 9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," which does not specify protective functions to be bypassed. The concern was that the licensee's setup could result in a higher probability of EDG loss during accident conditions. This item was left open pending review by the NRC program office.

Inspector Followup

Based on discussions with the NRC program office, the inspectors determined that the licensee's adherence to Regulatory Guide 1.9 was satisfactory.

E8.16 (Open) Unresolved Item 50-313/97201-14: Lack of Testing Unit 1 Molded Case Circuit Breakers

Background

The Updated Final Safety Analysis Report identified in Section 8.3 that the Class 1E electrical distribution system was designed to meet the requirements of IEEE Standard 308-1971, "IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations." Section 6.3 of this standard required periodic testing at scheduled intervals to demonstrate that components that were not exercised during normal operation of the station were operable. The standard further stated, "The specific tests and the frequency at which they are performed depend upon the specific components installed, their function, their environment, and the fact that they are in the maintenance program of the unit(s). Illustrative examples of tests are given in Table 2." Table 2 of the standard identified switchgear and one of the tests identified for illustrative purposes for that type of component was an overcurrent trip test.

In contrast to these requirements, the licensee did not have a program to perform periodic overcurrent trip testing of its Unit 1, Class 1E, molded-case circuit breakers.

The licensee developed a position paper that concluded that IEEE 308-1971 was not applicable to molded-case circuit breakers. This conclusion was based on an interpretation that Table 2 of the standard identified four tests for 600 V and below switchgear: operation test, mechanical inspection, overhaul, and overcurrent trip test.

Due to the physical construction of molded-case circuit breakers, mechanical inspection and circuit breaker overhaul were not possible. Therefore, the licensee concluded that the periodic tests in Table 2 were not applicable to molded-case circuit breakers but were rather intended for load center switchgear.

The inspectors considered the licensee's interpretation of Table 2 of IEEE 308-1971 to be in error, based on the following points:

- Table 2 identified illustrative periodic tests for Class 1E components.
- The identified tests were not an exhaustive or restrictive lists nor was Table 2 intended to exclude components for which it was not physically possible to perform all of the identified tests.
- The user of the standard was expected to evaluate which of the identified tests (or others) were appropriate for each component.

As a result of its operating experience assessments for NRC's Information Notices 92-51, "Misapplication and Inadequate Testing of Molded-Case Circuit Breakers" and 93-64, "Periodic Testing and Preventive Maintenance of Molded-Case Circuit Breakers," the licensee initiated a practice of performing electrical testing of molded-case circuit breakers in the receipt inspection process and performing mechanical exercising of the breakers on a periodic basis. However, they did not implement the practice of periodic electrical testing. Information Notice 93-64 referred to Electrical Power Research Institute Report NP-7410, "Breaker Maintenance." Volume 3 of this document, "Molded Case Circuit Breakers," recommended electrical testing of molded case circuit breakers on a 4- to 6-year periodicity.

Prior to the inspection documented in NRC Inspection Report 50-313/97-201, the licensee performed a self-assessment that identified that periodic electrical testing of molded-case circuit breakers was not performed. The licensee established a task group to determine if a testing program, replacement program, or a combination of both should be adopted.

Inspector Followup

The inspectors conducted interviews with personnel and reviewed the licensee's position paper on molded-case circuit breaker testing and plans for developing a periodic testing program for molded-case circuit breakers.

The licensee identified that they performed several activities related to safety-related molded-case circuit breakers to verify that they will function under design basis conditions and that the breakers are maintained according to vendor requirements.

These activities included: testing Unit 2 electrical penetration breakers in accordance with the Technical Specifications, testing all new breakers prior to installation, preventive maintenance inspection and cleaning of breakers and panels, manual cycling of some breakers, and performing periodic thermography inspections of 480 V molded-case circuit breakers. The licensee stated that these testing activities have historically resulted in very low failure rates.

The licensee informed the inspectors that they had undertaken a project to develop a periodic electrical testing program for molded-case circuit breakers. Specifically, the licensee stated that a program would be developed to test the Unit 1 containment electrical penetration breakers and the program would be implemented in the refueling outage scheduled for the spring of 1998. At the time of the inspection, the licensee had identified a sample population of 64 molded-case circuit breakers for inclusion in the program, of which, approximately 12 were scheduled for testing in the next refueling outage. Additionally, the licensee stated that it would categorize the remaining safety-related Unit 1 breakers by risk significance and develop and begin implementation of a testing program by the end of 1998. With respect to Unit 2, the licensee identified that safety-related molded-case circuit breakers that were not included in the Technical-Specification-required testing program would be evaluated for risk significance and a testing program would be developed and implemented by the Unit 2 refueling outage scheduled in 1999.

Based upon the previous record of satisfactory Technical Specification-required electrical testing experience of Unit 2 molded-case circuit breakers and other maintenance and testing activities the licensee has implemented, the inspectors did not consider the licensee's lack of a periodic testing program for Unit 1 molded-case circuit breakers an immediate safety concern. However, the licensee's practices will be reviewed further with the NRC's program office to determine if the licensee's reference to IEEE 308-1971 in its Updated Final Safety Analysis Report imposed a regulatory requirement to implement an electrical testing program for molded-case circuit breakers. This unresolved item was left open pending review of this issue by the NRC program office.

E8.17 (Closed) Unresolved Item 50-313/97202-15: Lack of Including all Safety-Related Fuses in Procedure

Background

The NRC identified that the licensee had not adequately resolved previous concerns related to incorrect electrical fuse installations. The licensee had established a like-for-like replacement policy for installed fuses but had physically verified only 4 percent of the fuses in Unit 1 to be correct in all aspects. The fuse control procedure required an engineering evaluation only for blown fuses or when replacement fuses were different from those installed. All other applications involved a like-for-like replacement without an engineering review. Under these conditions, an existing incorrect fuse would be replaced with a like fuse. The NRC considered the like-for-like fuse replacement policy to be inappropriate considering the small percentage of fuses validated.

Inspector Followup

In response to the NRC finding, the licensee proposed to add the following DC distribution equipment to the fuse control program by June 30, 1998: D01, D02, D15, D16, and D17 for Unit 1 and 2D01, 2D02, 2D26, 2D27, 2D41, 2D42, 2D51, and 2D52 for Unit 2. Also, as a long-term corrective action, the licensee stated that whenever a design change was made to a safety-related circuit protected by a fuse, a review of the fuse list would be performed by engineering to verify, correct, or add fuses to the procedure as necessary.

The licensee reviewed their condition report database for fuse-related discrepancies occurring since January 1, 1993. Seven condition reports during this period addressed wrong size or type of fuse. Only one of these condition reports affected Unit 1, and this fuse was installed in a nonsafety related circuit. One was in a nonsafety circuit common to both units and five affected Unit 2. Of these five, only three affected safety-related circuits. None affected circuit operability. Based on these facts, the licensee concluded that the fuse control program was being effectively managed. The inspectors determined that it was unlikely that a pervasive fuse control problem existed and, therefore, did not consider the matter to represent a safety concern.

E8.18 (Closed) Unresolved Item 50-313/97201-16: FSAR Discrepancies

Background

The NRC identified five discrepancies in the license's FSAR, mostly involving Table 7-11A, "R.G. 1.97 Post Accident Monitoring Variables." The discrepancies involved inconsistencies between the table and design documents concerning instrument ranges and display configurations. One additional discrepancy concerned a note in FSAR Section 14.2.2.1.3.1 stating that both steam generators would be isolated in the event of a steamline break; whereas, in actuality, only one steam generator would be isolated.

Inspector Followup

Prior to the identification of these discrepancies, the licensee was in the midst of a comprehensive FSAR review for both units. This program was described to the NRC in Letter CNRO-97/00010, "EOI Licensing Basis Assessment and UFSAR Review Initiatives," dated May 7, 1997. The licensee stated that this effort would resolve the generic concerns raised by this item since this review effort would identify the same types of errors discovered by the NRC. The specific errors were to be reviewed and corrected using the licensing document change process. These corrections were to be made effective in the 1998 submittal of the ANO Unit 1 FSAR.

The inspectors reviewed Letter CNRO-97/00010 and noted that it referred to a generic review of the Unit 1 and Unit 2 FSAR for accuracy. The inspectors reviewed a document entitled, "Safety Analysis Report Upgrade Project," that explained the scope of this effort, currently scheduled for completion in July 1998. Based on review of this material and

discussions with licensee personnel, the inspectors concluded that the project as planned would resolve the generic concerns related to this issue. At the time of this inspection, the licensee was approximately 50 percent complete with this project and had identified 895 potential discrepancies. Of these, 391 were strictly editorial, leaving 504 items having potential significance. However, the licensee stated that many items may subsequently be determined not to be FSAR errors since the reviewers have been instructed to document everything that might be a discrepancy.

The inspectors reviewed licensing Information Requests L97-0213, L97-0326, and L97-0310 and confirmed that the specific errors identified by the NRC were scheduled to be corrected in the next FSAR update.

The inspectors determined that the licensee's efforts were sufficient to correct the specific and generic concerns associated with this inspection finding.

10 CFR 50.71(e) states that the updated FSAR shall be revised to include the effects of all changes made to the facility or procedures as described in the FSAR. The licensee's failure to properly update the FSAR regarding the five discrepancies described above was considered a violation of 10 CFR 50.71(e). However, because of the favorable progress being made by the licensee's FSAR review team and the fact that a description of this project has been placed on the docket, and the NRC's view that the discrepancies identified by the NRC likely would have been identified by the licensee's program, the NRC is exercising discretion in accordance with Section VII.B.3 of the Enforcement Policy and is not taking formal enforcement action for this finding.

E8.19 (Closed) Unresolved Item 50-313/97201-17: Ineffective Resolution of Previous Corrective Actions

Background

During a plant walkdown, the NRC noted that two in-parallel check valves located in the steam supply piping to the turbine-driven emergency feedwater pump (P7A) were continually chattering because of steam leakage through downstream solenoid valves. The NRC's concern was that wear from this mechanical action could eventually result in failure of the check valves to perform their reverse-seating function.

Inspector Followup

The inspectors were informed that leakage through the solenoid valves had been a problem at ANO since at least 1988 when CR 1-88-0015 identified this problem. The licensee had performed maintenance on the valves on several occasions, but the leakage problem persisted. At some point, the licensee determined that the small amount of leakage was not detrimental to the system, since the check valves were not experiencing abnormal wear and the heating of the downstream piping was considered beneficial in reducing the potential for condensate-induced turbine overspeed events. Then, the licensee deferred any plan for modification. However, in October 1996, the licensee identified accelerated wearing of elastomeric surfaces on one of the

two solenoid valves in the EFW steam lines. As documented in Condition Report CR-1-96-0569, electricians found the O-ring on the operator of Solenoid Valve SV-2663 to be hard and brittle following its installation on July 14, 1992. The licensee's equipment qualification program stated that this O-ring was qualified for 60 years of service but did not consider seat leakage to be a normal condition for the establishment of qualified life. The licensee determined that the solenoid valve remained operable with the damaged O-ring. Based on the O-ring failure, the qualified life of the O-rings was being revised to be from 2 to 8 years depending on the service condition.

While the licensee considered a modification to the solenoid valves, they were also considering installation of motor-operated globe valves and relocating the valves closer to the turbine.

The check valves performed a safety-related function by isolating the piping from a faulted steam generator. The licensee stated that one of these valves would be disassembled and inspected during the next refueling outage in the Spring of 1998. In February 1995, the other check valve was inspected and found structurally sound with slight indications of scoring. During past tests of the check valve's reverse seating capability, the results have been acceptable.

The inspectors were informed that the check valves had chattered since their installation in 1986, even before the solenoid valves began to leak, because of the presence of a drain/strainer orifice that creates a constant steam demand of 1600 pounds-mass per hour. Since the check valves require 5000 pounds-mass per hour to remain constantly open, the solenoid valve leakage could actually reduce the chattering if it were of sufficient magnitude. However, the current chatter rate of the check valves was 10 taps per second, indicating that the solenoid leakage was well less than 3400 pounds-mass per hour.

The inspectors determined that the licensee had adequately ensured that EFW system operabililty had not been compromised by the solenoid leakage problem. The inspectors recognized that the check valve chattering was partially independent of the leakage problem and that the leakage had been observed to be beneficial for preventing condensation in the EFW turbine steam supply line. These facts mitigated any safety concerns related to this condition.

E8.20 <u>(Closed) Unresolved Item 50-313/97201-18: Lack of Seismic Support of OTSG Pressure Transmitters</u>

<u>(Closed) Unresolved Item 50-313/97201-19: Lack of Design Basis for Support of Steam Generator Instrument Sensing Lines</u>

<u>Background</u>

The NRC observed that a clip designed to restrain the isolation valve and sensing line for Steam Generator Pressure Transmitter PT-2667B was missing. The concern was that the line could fail during a seismic event, potentially affecting the operability of the steam

generator pressure control loop. The NRC also observed that the corresponding clips on the three other redundant pressure instrument channels in the same vicinity were loose. The licensee initiated Condition Report CR-1-97-0058 to determine the operability status and potential generic concerns. Before the end of the inspection, the licensee reported to the NRC that the as-found configuration was operable and that a walkdown of other instrument tubing restraints indicated that the mounting clip problem was not widespread.

In the Unit 1 EFW pump room, the NRC found a 10-foot unsupported span of instrument tubing for a pressure transmitter (PT/PI-2811). This is a safety-related sensing line that monitors the discharge pressure of the turbine-driven emergency feedwater pump. A maximum unsupported span of 2.5 feet was generally specified in the seismic drawing details. The licensee issued Condition Report CR-1-97-0074 and, subsequently, determined that the as-found configuration was operable. The licensee determined that a generic issue existed and elevated CR-1-97-0074 to a significant condition, requiring a root cause determination. The licensee performed a detailed walkdown of the EFW, Decay Heat/Low Pressure Injection, and High Pressure Injection systems and identified the following conditions:

* Several installations were undocumented or unanalyzed, such as tubing supports exceeding the 30-inch unsupported span and use of nonstandard instrument supports.

* Lack of design bases or criteria for determining instrument line slopes.

* Drawing discrepancies between drawings and the as-built condition.

* Missing or incorrect tags on instrument valves.

Based on these findings, the licensee expanded the scope to include other systems. For example, Condition Report CR-1-97-0087 was later opened to document inadequately supported emergency diesel generator instrument lines. In all, 37 instrument installations were found in Unit 1 that did not conform to either seismic or instrument installation requirements as defined in the standard drawing details. Design engineering determined that none of the 37 discrepancies created an inoperable condition.

Inspector Followup

The inspectors reviewed Root Cause Analysis Report CR-1-97-0074, "Instrument Tubing not Mounted in Accordance with Seismic Design Requirements," dated March 24, 1997.

The licensee identified the following root causes for this event:

(1) Inadequate training: Personnel involved in the installation and maintenance of instrument tubing were not adequately trained or sensitized to seismic and configurations requirements.

(2) Work practices: Personnel installing and maintaining instrument tubing either did not use required configuration documentation during work activities or did not follow these documents correctly.

(3) Design configuration and analysis: Design documentation and prints were inadequate or not easily accessible.

The licensee walked down 40 instrument installations in Unit 2 and found that the quality of the installations was much better than in Unit 1. Only four of the installations inspected had notable deficiencies involving damaged mounting clips or brackets resulting from removal and reinstallation to support work on nearby equipment.

The licensee performed the following corrective actions: (1) the individual discrepancies were repaired as necessary, (2) a letter was issued on April 4, 1997, to site groups reiterating instrument tubing requirements, and (3) design engineering reconfirmed the initial operability assessments from the tubing walkdowns. Additional long-term actions included the following: (1) develop a scope for additional walkdowns, (2) determine whether existing seismic tubing mounting requirements can be relaxed, (3) evaluate enhancements to design drawings associated with tubing, (4) evaluate the work planning processes that involve instrument tubing or the removal of tubing for interference purposes, and (5) provide training on seismic and configuration requirements for instrument tubing.

The inspectors discussed the status of the proposed long-term actions discussed above with the licensee. A calculation was in the final stages of review that redefined the acceptable support span lengths for various configurations. Once this calculation was complete, the licensee was planning to perform additional walkdowns of those safety-related components in Unit 1 that were potentially susceptible to instrument tubing discrepancies. This effort was to be completed by May 1998. The licensee was also planning to provide personnel training and to clarify drawings that contained confusing tubing support information.

The inspectors determined that the licensee's completed and proposed corrective actions (each of which were assigned completion due dates) were sufficient to address the issues raised by this item and considered the overall response to be strong, particularly considering that none of the identified discrepancies had been evaluated to affect operability.

10 CFR Part 50, Appendix B, Criterion III, "Design Control," states that the design control measures shall provide for verifying or checking the adequacy of the design.

The licensee's failure to provide measures to ensure the correct translation of seismic design requirements to instrument tubing installations and the maintenance of this configuration during plant operations in Unit 1 was identified as a third example of a violation of 10 CFR Part 50, Appendix B, Criterion III (50-313/9721-02).

Conclusions

The licensee's failure to provide measures to ensure the correct translation of seismic design requirements to instrument tubing installations and the maintenance of this configuration during plant operations in Unit 1 was identified as a violation.

E8.21 (Closed) Unresolved Item 50-313/97201-20: Inadequate Work Plan for the Control of Post-maintenance Testing

Background

The NRC identified a misaligned limit switch on Valve CV-4804, reactor building vent. The misalignment caused the actuator cam to drag on the limit switch arm toward the end of the valve stroke instead of riding on the roller bearing. Repeated operation in this configuration could have resulted in a faulty valve position indication. Although this particular application was not safety-related, there existed a concern that other safety-related applications may have a similar misalignment problem.

The licensee recognized that the post-modification plan and job order for this valve (00954697) did not include a post-maintenance test to verify proper operation of the limit switch cam assembly. The licensee reopened this job order and accompanying work plan with the intent to add steps to reconfigure and test the assembly. One other valve in Unit 1 (CV-1845, quench tank sample valve) had a similar mounting design, and a similar revision was scheduled for its work plan. The licensee performed a review for other similar limit switch configurations but did not identify any other of these configurations in either Units 1 or 2.

Inspector Followup

The inspectors reviewed the work plan revisions discussed above and observed that steps were added to verify that the limit switch rollers were in proper contact with the cams. The licensee considered that this verification in the work plan coupled with the existing post-maintenance test that verifies smooth motion and proper valve position light sequencing would be sufficient to ensure that the condition would not recur (i.e., no revision to the post-modification plan was considered necessary). The inspectors observed that the post-maintenance testing problem, in this instance, did not affect safety-related equipment and determined that the licensee had adequately addressed the specific and generic implications of this finding.

E8.22 (Closed) Unresolved Item 50-313/9721-21: Inadequate 50.59 Review Associated with the Removal of the Unit 1 BWST Vacuum Breaker and Followup Corrective Action

Background

This item involved an inadequate 10 CFR 50.59 safety evaluation performed by the licensee for a temporary covering placed over the borated water storage tank vacuum relief valve flange. The evaluation was inadequate, because it did not consider the air

flow restricting effects of a mesh screen that was an integral part of the temporary covering. The following is a summary of key events:

- On December 4, 1996, while Unit 1 was at full power, the licensee removed the 8-inch diameter vacuum relief valve from the borated water storage tank for a surveillance test. When the valve was removed, the licensee installed a white plastic bag over the 8-inch diameter open tank flange to act as a foreign material exclusion (FME) cover. Two slits, each approximately 8 inches long, were cut in the plastic cover to allow air flow. This work was done using Job Order 00952943. The job order stated that if suction for the tank was necessary, a supplemental vent path was provided by a 4-inch tank overflow line that was always open during plant operations.

- On December 5, 1996, the system engineer observed that the tank flange was covered with what he believed to be a solid FME cover and initiated Condition Report 1-96-0663 to document the potential inoperability of the borated water storage tank. Subsequently, the system engineer determined that the tank was operable since the FME cover contained two slits that would permit tank pressure and vacuum relief. The plastic bag was installed for approximately 30 hours. The licensee subsequently removed the plastic bag and installed a blank flange on the tank flange with a 1-inch gap between the flange and tank flange to permit pressure and vacuum relief. Around the 1-inch opening, a mesh screen was placed to prevent foreign material introduction into the tank.

- Condition Report 1-96-0663 also documented the discovery of a through-wall crack on the inlet flange of the removed vacuum relief valve and assessed the operability of the tank. The operability assessment did not address tank operability while the plastic bag was installed as an FME cover. The assessment only addressed the flange and 1-inch gap and stated that the blank flange with the 1-inch air gap installed over the tank flange would provide sufficient air movement to vent the tank during rapid drain downs. At this time, any air-flow restriction from the installed mesh screen was not considered.

- On December 12, 1996, the gap dimension was changed from 1 to 3-inches under Job Order 00952943. This was done to increase the venting capacity based on engineering judgement that the 1-inch gap may have been marginal.

- On January 22, 1997, Significant Condition Report 1-97-0019 was initiated and identified that a safety evaluation was not performed for the plastic bag installed over the tank flange for approximately 30 hours. In addition, the condition report identified that the procedure for testing the valve (Procedure 1306.034, "Testing of Unit 1 Pressure Vacuum Relief Valve PSV-1617, PSV-2423, and PSV-1412," Revision 2) did not allow removal of the vacuum relief valve during power

operations and that the safety evaluation for the procedure did not include consideration for valve removal at power. The condition report included an operability assessment for the tank when the plastic bag was installed on the tank flange. The licensee stated that the tank was operable based on engineering judgement that the plastic bag would break before an internal vacuum would cause damage to the tank.

- On February 3, 1997, the licensee issued Condition Report CR-1-97-0031 to address the previously made assumption that the 4-inch overflow line would, by itself, provide sufficient vacuum relief. The conclusion of this effort was that, contrary to prior judgement, the 4-inch line was not large enough to prevent the formation of an internal vacuum in the tank during rapid draindown.

- On February 11, 1997, the licensee prepared Temporary Alteration 97-1-001 that documented the removal of the vacuum relief valve and the installation of the blank flange with a 3-inch gap over the tank flange. The temporary alteration was prepared approximately 2 months after the actual installation. Temporary alterations were not issued for the previous two temporary coverings (plastic bag and 1-inch gap flange connection). Temporary Alteration 97-1-001, which contained the first 10 CFR 50.59 evaluation performed in association with these events, concluded that the borated water storage tank was operable with the 3-inch gap blank flange installation.

- On March 4, 1997, in response to concerns from the NRC, Condition Report 1-96-0663 was amended to include a discussion of the air flow blockage caused by the mesh screen over the 1-inch gap. After considering the screen restriction, the licensee concluded that the borated water storage tank was still operable under this configuration.

Inspector Followup

The inspectors reviewed the documents referenced above and had discussions with licensee personnel. In review of the facts, the inspectors identified several concerns in addition to those addressed in the previous inspection, as discussed below.

Procedure Issues

The inspectors found, in agreement with Significant Condition Report CR-1-97-0019, that the licensee's use of Procedure 1306.034 during power operations was improper. This procedure was originally written for use during refueling outage conditions as indicated by paragraph 4.3.1 of the procedure, which stated that inspection of the valve would be performed each refueling outage. The 10 CFR 50.59 evaluation for the procedure did not address performing maintenance or testing of the vacuum relief valve while the plant was at power. Also, the procedure did not address temporary provisions for vacuum relief, which would be an essential element for online maintenance of the vacuum relief valve. Although one of the corrective actions from the condition report was for the

licensee to revise Procedure 1306.034 to allow valve removal and maintenance of the valve while at power, this revision had not been completed at the time of this inspection.

10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. The licensee's failure to comply with the plant condition prerequisites of Procedure 1306.034 was identified as an apparent violation (50-313/9721-03).

Design Control Issues

The inspectors determined that the licensee had performed temporary changes to the borated water storage tank without the use of proper design control measures. The licensee removed the tank vacuum relief valve and sequentially installed three temporary covering configurations, each of which constituted a change to a safety-related component, without use of the temporary alteration procedure (a temporary alteration was eventually prepared for the third configuration, but this was issued approximately 2 months after installation).

The inspectors reviewed Procedure 1000.028, "Control of Temporary Alterations," Revision 20. Attachment 1 to this procedure listed the criteria for requiring a temporary alteration, which included a situation where a design change was made to safety-related equipment without the removal of the equipment from service. Using this criteria, the inspectors concluded that, because the borated water storage tank was not taken out of service for these evolutions, a temporary alteration was required. The use of a temporary alteration was intended to ensure that plant changes were subjected to design control measures commensurate with the original design.

10 CFR Part 50, Appendix B, Criterion III, requires those design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. The licensee's failure to provide design control measures commensurate with initial design for three temporary changes made to the borated water storage tank vacuum relief valve was identified as an apparent violation (50-313/9721-04).

Operability Issues

The plastic bag was attached to the tank flange in a manner such that the 8-inch ventilation slits may not have been pulled into the opening during an event (e.g., a safety injection event due to a large break loss of coolant accident that would cause a rapid tank draindown). Therefore, the venting function of the line could have been defeated. This would have left the 4-inch overflow line as the only venting source for the tank, which, according to the licensee's calculations, was not adequately sized to prevent a vacuum from forming in the tank during a rapid tank draindown. Consequently, the inspectors questioned the operability of the borated water storage tank during the 30

hours that the plastic bag was attached to the vent flange. Because of a lack of redundancy, an inoperable borated water storage tank would render the entire emergency core cooling system inoperable. The licensee performed several tests and calculations in an attempt to support the original engineering judgement that the tank was operable in this configuration. Following the onsite inspection period, the licensee provided the following evaluations and calculations for NRC review:

- "BWST [borated water storage tank] Past Operability," internal memorandum from Don Phillips to Charles Tyrone, dated November 25, 1997.

- Calculation 97-E-02100-01, "BWST Venting Capability Through 4-inch Overflow," Revision 0, December 3, 1997.

- Calculation 97-E-0211-01, "BWST Level Analysis," Revision 1, December 4, 1997.

- Calculation 97-E-0209-01, "Evaluation of BWST Tank T-3 for Potential Vacuum Load in Support of Operability Determination associated with CR-1-96-0663," Revision 0, December 8, 1997.

The licensee concluded that borated water storage tank was operable during the period of time the plastic bag was attached to the vacuum relief valve flange. However, the analyses clearly demonstrated that the borated water storage tank was in a degraded condition and could have plastically deformed above the water level in the tank in the event of a large or medium break loss of coolant accident. This deformation, according to the licensee's evaluation, would not have prevented the operation of the emergency core cooling system as designed. Also, the effect of the internal vacuum on the tank level instrumentation could have caused an early transfer of emergency core cooling water system suction from the borated water storage tank to the reactor building sump. This early transfer would have reduced the net positive suction head margins for the emergency core cooling system pumps. The licensee determined that this would not have rendered any of these pumps inoperable.

10 CFR 50.59 Issues

The inspectors determined that the licensee had failed to evaluate whether temporary changes made to the borated water storage tank had created an unreviewed safety question.

10 CFR 50.59, "Changes, Tests and Experiments," permits the licensee to make changes to the facility and to procedures as described in the Safety Analysis Report without prior Commission approval provided the change does not involve a change to the Technical Specification and does not involve an unreviewed safety question. A

proposed change, test, or experiment shall be deemed to involve an unreviewed safety question if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created. The licensee shall maintain records of changes in the facility, and these records must include a written safety evaluation that provides the bases for the determination that the change does not involve an unreviewed safety question.

The inspectors identified the following two examples where the licensee made changes to the borated water storage tank vacuum relief line without adequately evaluating whether an unreviewed safety question existed:

- The licensee removed the existing vacuum relief valve and covered the tank flange with a plastic bag taped to the tank. In the absence of a safety evaluation, the licensee used engineering judgement to consider the tank operable under this configuration. However, when questioned, the licensee had to perform extensive testing and analysis before concluding that the original engineering assumption was correct and that the tank was operable. The installation of the plastic bag appeared to create the possibility for an accident or a malfunction of a different type than any evaluated previously in the Safety Analysis Report and may have created an unreviewed safety question.

- The licensee installed a blind flange separated by a 1-inch gap from the tank flange and installed a foreign material exclusion screen over the gap. The unreviewed safety question evaluation of this configuration failed to consider the air flow restriction created by the foreign material exclusion screen.

The failure to properly determine whether an unreviewed safety question existed as required by 10 CFR 50.59 was identified as an apparent violation (50-313/9721-05).

Conclusion

Three apparent violations were identified associated with temporary coverings placed over the borated water storage tank vacuum relief valve flange, when this valve was removed on December 4, 1996, for testing and maintenance. The apparent violations involved the failure to perform proper safety evaluations as described in 10 CFR 50.59, the failure to perform the vacuum relief valve testing under the plant conditions as prescribed in the test procedure in accordance with the requirements of 10 CFR Part 50, Appendix B, Criterion V, and the failure to utilize the correct design control processes (temporary alterations) for the temporary cover configurations placed over the valve flange in accordance with 10 CFR Part 50, Appendix B, Criterion III.

E8.23 (Closed) Unresolved Item 50-313/97201-22: Revised Calculations Associated with BWST Vortexing and Pump NPSH

Background

The licensee prepared an engineering report, 93-R-1002-01, "ANO-1 BWST Outlet Vortex Suppressor," dated February 5, 1993, to examine the potential for vortexing in the BWST resulting in a loss of net positive suction head (NPSH) for the emergency core cooling pumps. The conclusion of the report was that the ECCS pumps would have sufficient NPSH. Within this report, the licensee identified that it had failed to consider the effect of instrument error in the vortexing calculation. This meant that the actual level in the tank at the time of switchover from the BWST injection mode to the recirculation mode may have been less than the nominal value, bringing the hydrodynamic system closer to a state of cavitation and air entrainment than had been originally assumed. The licensee issued Condition Report CR-1-97-0039 to investigate this condition. Within this report, the licensee concluded that the ECCS system would have remained operable. The licensee concluded that only the high pressure injection pumps would be susceptible to an NPSH problem but only at low tank levels near the switchover point to the recirculation mode. At this time in the accident, the high pressure injection (HPI) pumps could be shut down because the low pressure injection pumps alone could provide adequate core cooling. Within CR-1-97-0039, the licensee also was to complete Revision 1 to Engineering Report 93-R-1002-01 and to determine whether other calculations may have been affected by a similar failure to consider instrument error.

Inspector Followup

The inspectors reviewed the documents listed above and discussed the issue with licensee engineers. The inspectors determined that this instrument error was only reflected in the vortexing calculation and did not affect actual instrument readings.

10 CFR Part 50, Criterion III, "Design Control," states that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews.

The licensee's failure to check properly the adequacy of Engineering Report 93-R-1002-01 was identified as a fourth example of a violation of 10 CFR Part 50, Appendix B, Criterion III (50-313/9721-02).

Conclusions

The licensee's failure to check the adequacy of an engineering report properly that failed to account for instrument error in a calculation of vortexing in the borated water storage tank was identified as a violation.

E8.24 (Closed) Unresolved Item 50-313/97202-23: Inadequate 10 CFR 50.59 Evaluation Associated With BWST Releases

Background

This issue involved a problem with the 10 CFR 50.59 safety evaluation performed for Temporary Alteration 97-1-001, which is discussed in Section E8.23 of this report. In assessing the potential post-loss of coolant accident (LOCA) radioactive release through the 3-inch gap foreign material exclusion (FME) covering on the flange of the borated water storage tank (BWST) vent, the licensee addressed only offsite exposure and did not address control room exposure, which could be considerably more significant. Also, the safety evaluation did not refer to the licensee's response to Information Notice 91-56, "Potential Radioactivity Leakage to Tank Vented to Atmosphere."

Inspector Followup

The licensee revised the 10 CFR 50.59 evaluation for Temporary Alteration 97-1-001 to include a discussion of control room dose resulting from potential backleakage to the BWST. In the evaluation, the licensee stated that no accident consequence analysis takes credit for holdup or filtering of radioactive releases to the BWST, and, therefore, back leakage to the BWST is beyond the licensing basis. Based on recent test results of the BWST supply and return valves, the licensee stated (in the revised safety evaluation) that little or no backleakage would be expected to occur post-LOCA. Consequently, the licensee determined that there would be no change to the estimated control room dosage as a result of the temporary alteration.

The licensee reviewed its response to Information Notice 91-56 and determined that the conclusions reached as a result of Temporary Alteration 97-1-001 were not affected by this information notice.

The inspectors considered the lack of reference to control room dose in the original 10 CFR 50.59 evaluation to be an oversight, but observed that two mitigating factors existed. First, the original evaluation did not quantify a release rate based on the assumption that the release would be negligible. Had the release been quantifiable, an assessment of control room dose would have been necessary. Also, the fact that no credit was taken in the licensee's accident analysis for containment of BWST radioactive leakage relegated the issue to a beyond-licensing basis status. The inspectors considered the licensee's response to this issue to be adequate.

## IV. Plant Support

### F2 Status of Fire Protection Facilities and Equipment

a. Inspection Scope (92903)

During a walkdown of the Unit 1 emergency feedwater system pump room, which was performed in support of the review of various items discussed in Section E, the inspectors inspected the condition of the electrical cable fire barrier material.

b. Observations and Findings

The inspectors identified a deficiency in the fire wrap for Conduit EJ2029, which provided protection for the control circuitry for the turbine-driven emergency feedwater pump. On a vertical run of the conduit, a collar joining two sections of fire wrap had loosened and slid down. This resulted in about a 2-inch portion of the conduit being exposed.

The inspectors reported the condition to the licensee and showed the deficiency to a fire protection engineer and an operator. The licensee agreed that the as-found condition was unacceptable and immediately declared the fire barrier inoperable. In accordance with the requirements of its fire protection program, the licensee verified the operability of smoke detection equipment and control room alarm capability and established an hourly fire watch. Condition Report 1-97-0313 was initiated to document the item and Job Request 928623 was initiated to repair the fire wrap.

The inspectors asked the licensee whether there was a program for inspecting fire barriers and whether the fire wrap in the emergency feedwater pump room was included within the scope of the program. The licensee provided the inspectors with Procedure 1307.062, "Unit 1, 1-Hour Cable Fire Wrap Inspection," Revision 0. The inspectors reviewed the procedure, discussed fire wrap inspection techniques and the process of dispositioning inspection observations with licensee personnel, and verified that inspection of the pump room was included within the scope of the procedure. The inspection of the subject fire wrap was last performed with satisfactory results in July 1997. The procedure was required to be performed every 6 months. At the time of the inspection, no explanation was available for the cause of the deficiency. The licensee determined that a noncompliance did not exist, because the surveillance interval had not been exceeded and the damage could reasonably be assumed to have occurred following the most recent inspection.

The inspectors considered the licensee's actions on this matter to have been appropriate.

c. Conclusions

The inspectors identified a fire barrier deficiency and the licensee took prompt compensatory actions as required.

## V. Management Meetings

### XI Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management by telephone on January 6, 1998. The licensee acknowledged the findings presented. During this meeting, the licensee stated that it will expand its definition of margin of safety as it pertains to the performance of safety evaluations performed under 10 CFR 50.59. This issue is further discussed in Section E2.1 of this report.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. The licensee stated that one document reviewed by the inspectors contained proprietary information. The proprietary information from this document was not discussed within this inspection report.

## ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Ashley, Licensing Supervisor
E. Blackard, Design Engineer
M. Cooper, Licensing Specialist
G. Dobbs, Electrical and Instrumentation and Control Supervisor
J. Howell, Design Engineer
R. Hutchinson, Vice President, Arkansas Nuclear One
R. Lane, Director, Design Engineering
D. Mims, Licensing Director
D. Phillips, Nuclear Steam Supply System Supervisor
J. Richardson, Design Engineer
C. Tyrone, Director of Design Engineering
T. Waldo, Supervisor, Modifications
C. Zimmerman, Plant Manager, Unit 1

NRC

J. Melfi, Resident Inspector

## INSPECTION PROCEDURES USED

| | |
|---|---|
| 37001 | 10 CFR 50.59 Safety Evaluations |
| 92903 | Followup of Engineering Issues |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 50-313/9721-01 | VIO | Failure to Incorporate EFW Flow Limits into Procedures (Section E8.7) |
| 50-313/9721-02 | VIO | Design Calculation Deficiencies (Sections E8.8, E8.11, E8.20, and E8.23) |
| 50-313/9721-03 | APV | Failure to Perform BWST Vacuum Relief Valve Testing Procedure Under Proper Plant Conditions (Section E8.22) |
| 50-313/9721-04 | APV | Failure to Implement Temporary Alterations for BWST Vacuum Relief Valve Removal (Section E8.22) |

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|---|---|---|
| 50-313/9721-05 | APV | Failure to Perform 10 CFR 50.59 Evaluations for BWST Vacuum Relief Valve Removal (Section E8.22) |

Closed

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|---|---|---|
| 50-313/9627-03 | VIO | Failure to Notify NRC Within 1 hour of Declaration of NUE |
| 50-313/96-009 | LER | Fire in Reactor Building During Heatup Resulted from Cracked Weld in Oil Line |
| 50-313; -368/96512-01013 | EA | Inadequate Lube Oil Collection Systems for Reactor Coolant Pumps |
| 50-313/96512-01023, and 01033 | EA | Failure to Identify and Take Prompt Action to Lube Oil Leakage |
| 50-313/97201-01 | IFI | Licensee's Actions to Revise the Technical Specification Bases for the Minimum Water Volume in the Condensate Storage Tank |
| 50-313/97201-02 | URI | EFW Flow Rates Exceeding B&W Flow Rates |
| 50-313/97201-03 | URI | EFW Piping Configuration Differences |
| 50-313/97201-05 | URI | Evaluation of EFWP Pump Room Environment |
| 50-313/97201-06 | URI | Drawing and Calculation Revisions |
| 50-313/97201-07 | IFI | Modification Work Procedure Revisions |
| 50-313/97201-08 | IFI | Additional Vendor Information Necessary for Revised Cable Pulling Calculation |
| 50-313/97201-11 | IFI | Verification of Calculation Changes Associated With DC Batteries |
| 50-313/97201-12 | IFI | Review of Current Design Basis for Bypassing EDG Protective Trip |
| 50-313/97201-15 | URI | Lack of Including all Safety-Related Fuses in Procedure |
| 50-313/97201-16 | URI | FSAR Discrepancies |
| 50-313/97201-17 | URI | Ineffective Resolution of Previous Corrective Actions |
| 50-313/97201-18 | URI | Lack of Seismic Support of OTSG Pressure Transmitters |
| 50-313/97201-19 | URI | Lack of Design Basis for Support of OTSG Instrument Sensing Lines |
| 50-313/97201-20 | URI | Inadequate Work Plan for the Control of Post-Maintenance Testing |
| 50-313/97201-21 | URI | Inadequate 50.59 Review Associated with BWST Breaker |

| | | |
|---|---|---|
| 50-313/97201-22 | URI | Revised Calculations Associated with Vortexing and Pump NPSH |
| 50-313/97201-23 | URI | Inadequate 50.59 Evaluation Associated with BWST Releases |

Discussed

| | | |
|---|---|---|
| 50-313/97201-04 | URI | Inadequate Piping Pressure and Temperature Specifications |
| 50-313; -368/9623-01 | URI | Consideration of Multiple Hot Short Actuations |
| 50-313/97201-14 | URI | Lack of Testing of Unit 1 Molded Case Circuit Breakers |

## LIST OF ACRONYMS USED

| | |
|---|---|
| A/E | Architect/Engineer |
| BWST | Borated Water Storage Tank |
| CR | Condition Report |
| CRN | Calculation Revision Notice |
| DC | Direct Current |
| DCD | Design Configuration Documentation |
| DCIMS | Design Configuration Information Management System |
| EDG | Emergency Diesel Generator |
| EFW | Emergency Feedwater |
| FME | Foreign Material Exclusion |
| FSAR | Final Safety Analysis Report |
| MOV | Motor-operated valve |
| NPSH | Net Positive Suction Head |
| QAT | Quality Assurance Team |
| SIMS | Safety Information Management System |
| ULD | Upper Level Document |
| WBGT | Wet Bulb Globe Temperature |

## ATTACHMENT 2

### LIST OF DOCUMENTS REVIEWED FOR ITEM 50-313/97-201-06

**CALCULATIONS**

Calculation A-86, "Items Discussed with Daleas R.S. of CE MSLB, ECCS and LOCA," Revision 0.

Calculation Quality Action Team Final Report, September, 1996.

Calculation 32-1229968-00, "Final Draft, ANO AFW Capacity - Phase 1".

Calculation 32-5000450-00, "ANO AFW Capacity, Phase 2," Revision 0.

Calculation 80-D-1083A-02, "EFIC DC Valve Torque Calculation Under Reduced Voltage Condition," Revision 1.

Calculation 80-D-2085-01, "Correlation of Teletector to Activity Release," Revision 0.

Calculation 80-D-2085-02, "Provide Correlation Between MR/hr Reading Detector Activity Steam Line Result Table Method," Revision 0

Calculation 85-D-1042-03, "AMSAC Instr Loop Errors & Setpoints," Revision 5.

Calculation 85-E-0053-20, "Combustible Loading Calc for Fire Area G," Revision 29.

Calculation 86-E-0002-01, "EDG 1 & 2 Load Study," Revision 8.

Calculation 86-E-0020-01, "IE Station Battery 2D11 Duty Cycle & Sizing Calc," Revision 7.

Calculation 87-E-0026-09, "EFW Pump Room Temperature Profile," Revision 0.

Calculation 88-E-0086-01, "IE Bulletin 88-04 Review for P7A and P7B Minimum Flow Evaluation," Revision 0.

Calculation 89-E-0029-02, "Cal Factor Calc ANO-2 MS Line Rad Monitor," Revision 0.

Calculation 89-E-0102-01, "Terminal Voltage Calc for DC MOV," Revision 8.

Calculation 89-E-0144-01, "EDG Loading for 2A3 & 2A4 Bus," Revision 3.

Calculation 90-E-0116-01, "ANO-2 EOP Setpoint Document," Revision 4.

Calculation 91-E-0090-06, "Heat Load Determination for Post Accident for Rooms 95 98 99 100 104 109 110 & 149 for Post Accident Cooling," Revision 2.

Calculation 92-E-0021-01, "Emergency Duty Cycle and Battery Sizing Calculations," Revision 4

Calculation 92-E-0077-04, "Unit 1 EFW System Pump Performance Requirements," Revision 0.

Calculation 93-D-1008-05, "MOV Sizing Calc for CV-2620," Revision 0.

Calculation 93-R-1029-30, "ANO-1 SRR for Auxiliary Building HVAC System ABH V," Revision 0.

Calculation 94-D-5033-01, "Removal of Local Fire Trouble Horns from D21 Breaker 34," Revision 0.

Calculation 94-E-0061-01, "Tech Spec Allowable Outage Time (AOT) Utilizing Alternate AC EDG," Revision 0.

Calculation 95-D-1004-03, "DCP 95-1004 Load Addition to RC1," Revision 0.

Calculation 95-D-1011-01, "EDG Load Add for DCP 95-1011," Revision 1.

Calculation 95-D-7064-01, "Vital AC Panels & Instrument AC Panels," Revision 0.

Calculation 95-D-7072-01, "D01 Battery Loading Amendment," Revision 0.

Calculation 96-E-0065-01, "Unit 1 Safety Survivability Eval Under Stall Thrust/Torque," Revision 1.

Calculation 96-R-0002-02, "Generic Letter 89-10 Program Valves Limit Switch Setpoint," Revision 1

**CONDITION REPORTS**

CR-C-96-0060, re Calculation Room Control and Discrepancies, March 22, 1997.

CR-C-97-0053, re Calculation 92-D-2001-49 Approval, February 11, 1997.

CR-C-97-0058, re Upper Level Document Discrepancies, February 13, 1997.

CR-C-97-0059, re Design Configuration Documentation Discrepancies, February 13, 1997.

CR-1-96-0184, re Seismic Qualification Calculations, May 23, 1996.

CR-1-97-0017, re Accuracy Analysis Calculations for EFW Flow Transmitters, January 21, 1997.

CR-2-97-0304, re Discrepancy in Instrument Setpoint Calculations 2LT-5659-1 and 2LT-5668-2, May 28, 1997

**DESIGN CHANGES**

Design Change Package 94-1003, "Feedwater (MFW Venturi Replacement)," Revision 2

Design Change Package 95-2001, "Control Room Emergency Chiller Installation - Air Conditioning Modification," Revision 1

**ENGINEERING STANDARDS**

Engineering Standard GES-39, "Calculation Numbering Schemes and Codes," Revision 2

**LICENSING INFORMATION REQUESTS**

LIR L97-0035, "Action Items from ANO 50.54(f) Design Bases Assessment: Maintenance of Calculations [1.1, 1.2],".

LIR L97-0037, "Action Items from ANO 50.54(f) Design Bases Assessment: DCIMS Improvements [1.6],".

LIR L97-0201, "NRC AE Design Basis Inspection," July 31, 1997

LIR L97-0202, "NRC AE Design Basis Inspection," July 31, 1997.

LIR L97-0289, "Response to IR 97-202 AE Design Basis Inspection," September 26, 1997

LIR L97-0290, "Response to IR 97-201 AE Design Basis Inspection," September 26, 1997

**PROCEDURES**

Procedure OP-5010-015, "Engineering Calculations," Revision 1 PC1.

Procedure OP-5010-019, "Computer Software Control," Revision 1 PC1

**MISCELLANEOUS**

Memorandum ANO-97-00225, M. Stroud to ANO Engineering Personnel, "Reminder of Design Considerations for Possible Impact to Battery and EDG Loading Calculations," March 3, 1997

Design Configuration Documentation Discrepancies 2M7-07-01,02,03,07,08,09,12,14.

Design Configuration Documentation Discrepancies RBS 02-05,06