UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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April 21, 2003

Craig G. Anderson, Vice President,
Operations
Arkansas Nuclear One
Entergy Operations, Inc.
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Russellville, Arkansas 72801-0967

SUBJECT: ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 - NRC INTEGRATED INSPECTION REPORT 50-313/03-02; 50-368/03-02

Dear Mr. Anderson:

On March 22, 2003, the NRC completed an inspection at your Arkansas Nuclear One, Units 1 and 2, facility. The enclosed report documents the inspection findings, which were discussed with you and other members of your staff on April 3, 2003, and as described in Section 4OA6.

The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified issues that were evaluated under the risk significance determination process as having very low safety significance (Green) as well as one issue that required evaluation using our traditional enforcement process. The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Arkansas Nuclear One facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch D
Division of Reactor Projects

Dockets: 50-313
50-368
Licenses: DPR-51
NPF-6

Enclosure:
NRC Inspection Report
50-313/03-02; 50-368/03-02

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RIV:RI:DRP/D	SRI:DRP/D	PE:DRP/D	SPE:DRP/D	C:DRS/PSB
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T - LJSmith	***T - LJSmith***	***E - LJSmith***	***E - LJSmith***	***MPShannon for***
4/21/03	4/21/03	4/21/03	4/21/03	4/21/03
C:DRP/D				
LJSmith				
/RA/				
4/21/03				

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-313, 50-368
Licenses:	DPR-51, NPF-6
Report No:	50-313/03-02; 50-368/03-02
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	December 29, 2002, through March 22, 2003
Inspectors:	R. Bywater, P.E., Senior Resident Inspector J. Clark, Senior Project Engineer, Project Branch D R. Lantz, Senior Emergency Preparedness Inspector L. Ricketson, P.E., Senior Health Physicist K. Weaver, Resident Inspector
Approved By:	Linda Joy Smith, Chief, Project Branch D Division of Reactor Projects
Attachment 1:	Supplemental Information
Attachment 2:	ANO Paper, "Secondary System Pressure Boundary as an Extension of Containment Liner"

SUMMARY OF FINDINGS

Arkansas Nuclear One, Units 1 and 2
NRC Inspection Report 50-313/03-02; 50-368/03-02

IR05000313-03-02, IR05000368-03-02; Entergy Operations, Inc.; 12/29/02 - 03/22/03; Arkansas Nuclear One, Units 1 and 2; Evaluations of Changes, Tests, or Experiments; Temporary Plant Modifications; ALARA Planning and Controls.

The inspection was conducted by the resident inspectors, one senior project engineer, a senior health physicist, and a senior emergency preparedness inspector. The inspection identified three findings. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified Findings

Cornerstone: Mitigating Systems

Green. The licensee did not properly evaluate a temporary alteration that was performed when a door separating a safety-related switchgear room from the turbine building was removed for maintenance. As a result, the impact of a potential high energy line break on equipment needed to mitigate the event was not identified or evaluated by an engineering evaluation. Failure to perform an engineering evaluation to support this temporary alteration was a violation of Unit 1 Technical Specification 5.4.1.a. This violation is being treated as a noncited violation (NCV) consistent with Section VI.A in the Enforcement Policy.

The safety significance of this issue was determined to be very low since this issue screened as Green during a Phase 1 SDP assessment, because the finding did not result in equipment becoming incapable of performing its function in the case of a design basis accident. The issue was considered to be more than minor because it affected the mitigating systems cornerstone objective for design control and modifications because the ability to mitigate the consequences of a high energy line break would have been affected if the finding had affected more than one train of equipment (Section 1R23).

Cornerstone: Barrier Integrity

Severity Level IV. The inspectors identified a noncited violation of 10 CFR 50.59 because the licensee failed to identify that changes made to the Units 1 and 2 Updated Safety Analysis Reports required a license amendment request. These changes removed containment isolation valve controls for secondary system containment penetrations. The licensee initiated corrective action on March 28, 2003, to prepare a license amendment request to obtain NRC approval of the changes to the Updated Safety Analysis Reports.

This is an item for traditional enforcement because it involves an issue not appropriate for evaluation using the SDP. It involves a violation of 10 CFR 50.59, an issue which impacts NRC oversight ability. The issue is more than minor because it involves a programmatic issue affecting containment controls for all secondary system penetrations. It was considered to be a noncited Severity Level IV violation. Management review determined it was greater than minor because the change should have received NRC review prior to implementation. Redundant containment barrier (system piping) existed and the licensee entered this issue into its corrective action program (Section 1R02).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a noncited violation of Units 1 and 2 Technical Specifications 5.4.1.a and 6.8.1.a, respectively, because the licensee failed to follow procedural requirements. Specifically, the licensee failed to provide the reason radiation work permits and work activity dose estimates were revised as required by Procedure NMM RP-105, Revision 1, Section 5.8.

  The inspectors determined that this finding was associated with the Occupational Radiation Safety Cornerstone program and process attributes (ALARA planning/projected dose) and affected the objective of the cornerstone, which is to protect the worker from exposure to radiation. Therefore, the finding was greater than minor. The occurrence involved a failure to maintain or implement, to the extent practical, procedures needed to achieve occupational doses that were ALARA, which resulted in unplanned, unintended occupational collective dose for a work activity. Therefore, the safety significance of the finding was evaluated using the Occupational Radiation Safety SDP. However, because the licensee’s 3-year rolling average collective dose was not greater than 135 person-rem/unit, the finding had no more than very low safety significance.

B. Licensee-Identified Findings

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Unit 1 began the inspection period at approximately 100 percent power and remained at or near 100 percent throughout the inspection period.

Unit 2 began the inspection period at 100 percent power. On February 9, 2003, Unit 2 operators reduced reactor power to approximately 70 percent in support of engineering data collection for feedwater system testing. Unit 2 operators returned the unit to 100 percent power the same day.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity
[REACTOR-R]

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

On January 21-23, 2003, the inspectors walked down the Unit 2 safety-related battery rooms to verify that compensatory measures were in place for the cold weather conditions and to verify that the battery room temperature was sufficient to maintain the electrolyte temperature above the Technical Specification 4.8.3.b.3 limit of 60°F. The inspectors reviewed Unit 2 Procedure 2106.032, "Unit Two Freeze Protection Guide," Revision 9, to determine all applicable cold weather protection requirements for the Unit 2 station batteries and verified that these measures were in place to protect the equipment.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Unit 2 Containment Isolation Valve Designation Change

a. Inspection Scope

During a routine review of plant status for Unit 2, on or about December 21, 2001, to January 16, 2002, the inspectors noted that Unit 2 Steam Generator B Sample Valve 2CV-5859-2 had been de-energized in the open position and backseated due to a packing leak, which resulted in a Unit 2 control room panel alarm of "CIAS inop." This valve is located directly outside containment Penetration 2P7 and receives an engineered safety feature actuation signal to close on a containment isolation actuation signal (CIAS) on high containment pressure. With the valve de-energized open, it would not perform its function to close on a CIAS.

The inspectors found that Technical Specification 3.6.3.1, "Containment Isolation Valves," limiting condition for operation was not entered for this valve, while it was in the open position and de-energized for 26 days (December 21, 2001, to January 16, 2002). This exceeded the action time requirements of the Technical Specifications. The licensee stated that the valve was no longer considered a containment isolation valve. The inspectors performed a historic review of the licensing basis for this valve to determine why Valve 2CV-5859-2 was no longer considered a containment isolation valve within the scope of the action requirements of Technical Specification 3.6.3.1.

b. Findings

(1) Introduction

As a result of an inadequate 10 CFR 50.59 screening, the licensee failed to perform a safety evaluation for:

- Changing the description of containment documented in the Updated Final Safety Analysis Report (UFSAR)

- Changing the description of General Design Criterion (GDC) 57, "Closed System Isolation Valves," applicability documented in the UFSAR

- Changing Procedure 1015.034, "Containment Penetration Administrative Controls," Revision 1, to indicate that Unit 2 Steam Generator B Sample Valve 2CV-5859-2 was not a containment isolation valve within the scope of Technical Specification 3.6.3.1

Several other secondary system containment penetration valves were also included in these changes. The failure to perform a safety evaluation for these changes to the Unit 2 UFSAR and Procedure 1015.034 was determined to be an example of a violation of 10 CFR 50.59.

(2) Description

Background

As part of this historical review, the inspectors reviewed the Unit 2 Preliminary Safety Analysis Report and the Unit 2 Final Safety Analysis Report. These documents were used by the Commission to perform the radiological safety review with respect to a decision concerning issuance of an operating license for Unit 2. The staff's review was documented in NUREG-0308, "Safety Evaluation Report, Arkansas Nuclear One, Unit 2," Supplement 1, dated November 1977. NUREG-0308, Supplement 1, states that, "the radiological safety review with respect to a decision concerning issuance of an operating license for ANO-2 has been based on the applicant's Final Safety Analysis Report (Amendment 20) and subsequent Amendments 21-43, all of which are available for review at the Nuclear Regulatory Commission's Public Document Room . . . ." The Safety Evaluation Report (SER) summarizes the results of the radiological safety review of Unit 2 performed by the staff. The SER further states that "The conclusions reached as a result of the evaluation of the applicants application to operate Unit 2 are presented in

Section 22.0 of the SER." NUREG-0308, "Containment Isolation Systems," Supplement 1, Section 6.2.4, states that "The containment isolation system is designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided so that no single valve or piping failure can result in loss of containment integrity. Containment building penetration piping up to and including the external isolation valve is designed as seismic Category I equipment and is protected against missiles which could be generated under accident conditions. Containment isolation will occur automatically upon receipt of a containment isolation actuation signal of high containment pressure (5 pounds per square inch gauge.) All fluid penetrations not required for operation of the engineered safety features equipment will be isolated. Remotely operated isolation valves have been provided for those safety-related systems which will not be automatically isolated . . . ."

NUREG-0308, Supplement 1, further states that "we have reviewed the containment isolation system for conformance to General Design Criteria 54, 55, 56, and 57. We conclude that the system meets the General Design Criteria and is, therefore, acceptable."

As stated earlier, the staff documented in NUREG -0308, Supplement 1, that "the radiological safety review with respect to a decision concerning issuance of an operating license for ANO-2 has been based on the applicant's Final Safety Analysis Report (FSAR) (Amendment 20) and subsequent Amendments 21 through 43." FSAR Section 6.2.4 through Amendment 32, dated October 31, 1975, stated that "the containment isolation systems provide the means of isolating fluid systems that pass through containment penetrations so as to confine to the containment any radioactivity that may be released following a postulated accident. Unit-2 does not have a particular system for containment isolation; however, isolation is achieved by applying common criteria to penetrations in the various fluid systems, and by using a single parameter, containment pressure, to actuate the appropriate valves." FSAR, "Design Bases," Section 6.2.4.1, states that "the Design basis for the CIS is to minimize the release of radioactive material from the Containment by closing all fluid penetrations not serving accident consequence limiting systems, so that the site boundary thyroid and whole body doses from radioactive material escaping through the containment penetrations plus the doses from other sources during any postulated accident are within the limits of 10 CFR Part 100. The containment isolation systems are designed in accordance with 10 CFR Part 50, Appendix A, General Design Criteria 54, 55, 56 and 57, and meet the leak testing criteria of 10 CFR Part 50, Appendix J. The applicable criterion for each penetration is shown in Table 6.2-26." FSAR Table 6.2-26 and amendments used by the staff for the radiological safety review concerning issuance of an operating license for Unit 2 identified Containment Penetrations 2P1, 2P2, 2P3, 2P4, 2P7(SGA), 2P7(SGB), 2P32, 2P35, 2P64, and 2P65 as meeting the GDC 57 criterion. The above listed penetrations are all associated with the containment building secondary system piping penetrations.

Based on review of the Unit 2 FSAR, amendments documented in NUREG-0308, and the current UFSAR through Amendment 16, the inspectors found that Unit 2 Steam Generator B Sample Valve 2CV-5859-2 had always been previously identified and documented in the FSAR as a containment isolation valve, which met 10 CFR Part 50,

Appendix A, GDC 57, until a change was made in the UFSAR in Amendment 13. In addition, this valve had previously been identified as a containment isolation valve required to be operable in Unit 2 Technical Specification Table 3.6-1 (referenced by Technical Specification 3.6.3.1) with an isolation time of $\leq$20 seconds on a CIAS. The requirement that this valve be considered a containment isolation valve remained identified in Technical Specification Table 3.6-1 until License Amendment 154 was implemented.

The licensee requested and was granted approval in License Amendment 154 to remove Technical Specification Table 3.6-1 and replace Technical Specification 3.6.3.1 with the following: "Each containment isolation valve shall be OPERABLE." This was done in accordance with the guidance of Generic Letter 91-08, "Removal of Component Lists from Technical Specifications." The SER for Amendment 154 states that "the licensee confirmed that with the proposed changes, the new Technical Specifications applied to all valves classified as containment isolation valves by the plant licensing basis." The NRC staff found that the proposed changes to the Technical Specification were primarily an administrative change that did not alter the requirements set forth in the Technical Specifications. The NRC staff concluded that the changes would allow the licensee to make corrections and updates to the list of components for which the Technical Specification requirements applied, under the provision that these changes to plant procedures be controlled as specified in the Administrative Controls Section of the Technical Specifications. The SER also states that the Technical Specification Bases were amended to include reference to a new list of containment isolation valves that would be included in Procedure 2203.005, "Loss of Containment Integrity," and that the new list would be subject to the administrative control requirements of the Technical Specifications and 10 CFR 50.59. The inspectors noted that the new list of containment isolation valves was actually included in Procedure 1015.034, "Containment Penetration Administrative Control," Revision 0.

During review of NRC Generic Letter 91-08, the inspectors noted that the generic letter states that "Generally, the UFSAR identifies those valves that are classified as containment isolation valves. With this Technical Specification change, the limiting condition for operation (LCO), remedial actions, and surveillance requirements will apply for all valves that are classified as containment isolation valves by the plant licensing basis." The inspectors noted that Valve 2CV-5859-2 was identified as a GDC 57 containment isolation valve in the Unit 2 UFSAR, Table 6.2-26, "Containment Penetrations," and in Procedure 1015.034 at the time when License Amendment 154 was approved and incorporated into the Unit 2 Technical Specifications.

Description of Change to the UFSAR

During review of their containment penetrations, the licensee had identified that the licensing documents that specified containment isolation valves were not consistent and determined that this was caused by various personnel incorrectly interpreting the definition of the containment boundary. In 1995 and 1996, Procedure 1015.034, "Containment Penetration Administrative Control," Revision 0, and Unit 2 UFSAR, Table 6.2-26, were changed with Licensing Document Change Request 2-6-2-002 for UFSAR, Amendment 13. In this amendment, note 9 was added and the applicability of GDC 57 was deleted for 38 valves associated with the following Unit 2 secondary system containment penetrations:

Containment Penetration 2P1 Main Steam from S/G 2E24A - MSIS
Containment Penetration 2P2 Main Steam from S/G 2E24B - MSIS
Containment Penetration 2P3 Feedwater to S/G 2E 24A - MSIS
Containment Penetration 2P4 Feedwater to S/G 2E 24B - MSIS
Containment Penetration 2P7 S/G 2E24A Sample - CIAS
Containment Penetration 2P7 S/G 2E24B Sample - CIAS
Containment Penetration 2P32 S/G 2E24A Blow down - MSIS
Containment Penetration 2P35 EFW to S/G 2E24A - MSIS
Containment Penetration 2P64 S/G 2E4B Blow down - MSIS
Containment Penetration 2P65 EFW to S/G 2E24B - MSIS

Note 9 stated "These penetrations are associated with the secondary side of the steam generators and are not subject to GDC 57 since the containment barrier integrity is not breached during DBA LOCA conditions. The containment boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes, the outer surface of the lines emanating from the steam generator, and the outer surface of the steam generator above the bottom tube sheets. Valves associated with these penetrations are not containment isolation valves."

During discussions with the licensee’s staff, the inspectors were informed that this treatment of the secondary system as part of the containment boundary had always been the licensee's view of containment and that historical inconsistencies in the designation of secondary system containment valves could be explained, if this was understood. They also noted that their view was consistent with a 1972 Westinghouse document describing the containment boundary. See Attachment 2 for additional details regarding the licensee's perspective. The inspectors could not identify that the 1972 Westinghouse document was ever referenced in any licensing correspondence with the NRC that was applicable to Unit 2. The inspectors were also not able to find where the NRC staff had reviewed and accepted this alternative view of the containment boundary.

Screening Review

Approval for UFSAR, Amendment 13, was based on Engineering Report 93-R-0007-01, "Containment Penetration Design Summary Document," Revision 0. The inspectors reviewed Engineering Report 93-R-0007-01 and the associated 10 CFR 50.59 screening review. The following note was included in the Basis for Change for all of the secondary system containment penetration valves:

"ADMINISTRATIVE/EDITORIAL - Information is confusing in that it leaves the impression that these valves are Reactor Building Isolation valves when they are not. Deletion is consistent with note 9" (Unit 2 UFSAR, Table 6.2-26).

Because the changes made for all of the secondary system containment penetration valves were viewed as administrative/editorial, these changes met the licensee's criteria for changes specifically exempted from further 10 CFR 50.59 evaluation. This decision was in accordance with Procedure 1000.131, "10 CFR 50.59 Review Program," Attachment 1, which was, at that time, the licensee’s procedure governing 10 CFR 50.59 determinations/evaluations.

The inspectors reviewed containment descriptions in other sections of the initial FSAR and the NRC Safety Evaluation Report and did not find that the licensee’s view of containment had been previously documented. The inspectors determined that the change in definition of containment was not an administrative/editorial change because changes in the potential for increased fission product release during a steam generator tube rupture accident or other postulated accidents were not evaluated.

While the inspectors acknowledged that the original licensing basis for the containment boundary was in some cases silent and sometimes inconsistent, the inspectors noted the Unit 2 FSAR and amendments documented in NUREG-0308 included FSAR, Table 6.2-26, Amendment 23, which identified all the above secondary system Unit 2 containment penetrations and listed all of the secondary system valves as being GDC 57 containment isolation valves.

Based on the historical review, the inspectors concluded that on March 31, 1995, the licensee performed an inadequate 10 CFR 50.59 screening review documented in Evaluation FFN-95-166, which was approved by the Plant Safety Review Committee on August 31, 1995, for proposed changes to the Unit 2 UFSAR. As a result of this inadequate screening, the licensee had not performed a required safety evaluation.

10 CFR 50.59 was changed via publication in the "Federal Register" on October 4, 1999 (64 FR 53582). It is NRC policy to exercise enforcement discretion pursuant to Section VII.B.6 of the Enforcement Policy and not issue citations or document noncited violations against the old rule if the revised rule requirements were met.

Safety Evaluation Using the Revised Rule

At the inspectors' request, the licensee performed a safety evaluation of the change using criteria of the revised 10 CFR 50.59 rule for the Unit 2 Steam Generator B sample valve.

The licensee identified that the change would result in a very small increase in the consequences of an accident previously evaluated. The inspectors reviewed the guidance in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," and in NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, Section 4.3.3, and concluded that this increase in consequences would be minimal and, therefore, did not require NRC review for that reason.

The licensee did not identify that the change resulted in a small increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report, which should have received the NRC's approval prior to implementation. Using the guidance in NEI 96-07, Revision 1, Section 4.3.2, by definition, departure from GDC 57 results in more than a minimal increase in the likelihood of occurrence of a malfunction of a component important to safety. In addition, the probability of malfunction did increase when Valve 2CV-5859-2 was de-energized in the open position for 26 days (December 21, 2001, through January 16, 2002).

(3) Analysis

This finding is not suitable for evaluation using the Significance Determination Process (SDP). The finding is determined by management to be of very low safety significance. The inspectors reviewed an analysis performed by the licensee and found that the expected increase in consequences will meet the minimal standard described in NEI 96-07, Revision 1, Section 4.3.3. This finding is more than minor because the change should have received NRC's review prior to implementation.

(4) Enforcement

The inspectors evaluated the change in the UFSAR documented description of containment, the change in the UFSAR list of containment isolation valves, and the change in Procedure 1015.034 for compliance with 10 CFR 50.59. At the time that the 10 CFR 50.59 screening was performed in 1995, the rule required the licensee maintain records of changes in procedures described in the Safety Analysis Report and these records include a written safety evaluation which provides the basis for the determination that the change does not involve an unreviewed safety question. The licensee had not initially performed a safety evaluation for the change in the UFSAR documented containment description, for the deletion of multiple valves from the UFSAR list of containment isolation valves, and for the resultant deletion of Technical Specification applicability for these containment isolation valves in Procedure 1015.034. The failure to perform the safety evaluation was a violation of 10 CFR 50.59.

Under the previous 10 CFR 50.59 rule, the inadequate screening evaluation resulted in the licensee not identifying that the change would result in a small increase in the consequences of an accident previously evaluated and a small increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report. This constituted an unreviewed safety question which should have received NRC approval prior to implementation.

To evaluate conformance with the revised 10 CFR 50.59 rule, the inspectors reviewed the change in the UFSAR documented containment description, the deletion of multiple valves from the UFSAR list of containment isolation valves, and the resultant deletion of the Technical Specification applicability for these containment isolation valves in Procedure 1015.034 against the requirements of the new rule, using the guidance in Regulatory Guide 1.187, and NEI 96-07, Revision 1.

Under the revised rule, the increase in consequences was minimal. However, departure from GDC 57 does result in more than a minimal increase in the likelihood of occurrence of a malfunction of a component important to safety. Specifically, the probability of malfunction increased more than minimally when Unit 2 Steam Generator B Sample Valve 2CV-5859-2 was de-energized in the open position for 26 days (December 21, 2001, through January 16, 2002). Based on these findings, the inspectors concluded that a license amendment was also required by the revised rule.

Therefore, the NRC will not exercise enforcement discretion pursuant to Section VII.B.6 of the Enforcement Policy for the 1995 failure to perform a safety evaluation as required by 10 CFR 50.59.

The violation is not suitable for SDP evaluation, but has been reviewed by NRC management and is determined to be a Green finding of very low significance. The licensee entered this finding into its corrective action program as Condition Report ANO-2003-0242. This violation is being treated as an example of a noncited violation (50-313/0302-01; 50-368/0302-01) consistent with Section VI.A of the NRC Enforcement Policy.

.2 Unit 1 Containment Isolation Valve Designation Change

a. Inspection Scope

During the inspectors' review of the Unit 2 issue of deletion of GDC 57 requirements from certain Unit 2 containment isolation valves, the inspectors also found that multiple secondary system reactor building penetration valves which were previously identified as GDC 57 reactor building isolation valves were removed from the Unit 1 UFSAR in the same manner. The inspectors reviewed Engineering Report 93-R-0007-01, its associated 10 CFR 50.59 determination, and the Unit 1 UFSAR, Table 5-1, to determine if the licensee had appropriately performed these reviews.

b. Findings

(1) Introduction

The inspectors found that Engineering Report 93-R-007-01 and the associated 10 CFR 50.59 determination was used for changes to both the Units 1 and 2 UFSARs and containment penetration administrative control procedures. Therefore, the issue of the licensee’s departure from GDC 57 applicability for the steam generator secondary side reactor building/containment building penetration valves (which were previously identified as GDC 57 valves) from the Units 1 and 2 UFSARs and the Units 1 and 2 Technical Specification requirements as implemented by the respective Unit 1 and 2 containment penetration control procedures, is applicable to both Units 1 and 2.

As a result of an inadequate 10 CFR 50.59 screening, the licensee failed to perform a safety evaluation for:

- Changing the description of containment documented in the Unit 1 UFSAR
- Changing the description of GDC 57 applicability documented in the Unit 1 UFSAR

Several secondary system containment penetration valves were included in these changes. The failure to perform a safety evaluation for these changes to the Unit 1 UFSAR was determined to be an example of a violation of 10 CFR 50.59.

(2) Description

Background

As part of the inspectors' review for the Unit 1 issue, the inspectors found that a previous violation had been cited for Unit 1 concerning the failure to lock closed two reactor building isolation valves between the feedwater isolation valves and the containment in accordance with Technical Specification 3.6.5. This violation was documented in 1994 in NRC Inspection Report 50-313/94-07; 50-368/94-07. The subject valves were secondary system reactor building penetration feedwater line vent and drain Valves FW-1038 and FW-1049. They are located in secondary system penetrations that were affected by the licensee's UFSAR change. The inspectors reviewed the licensee response and corrective actions associated with the violation to understand the licensee's interpretation of the definition of containment at that time, just prior to the deletion of GDC 57 applicability for secondary system penetrations.

In the licensee response to the violation, dated December 16, 1994, the licensee stated:

"The identified concern is that main feedwater vent valves FW-1038 and FW-1049 were not being maintained as locked valves. Implied in the violation is that General Design Criterion (GDC) 57, Closed System Isolation Valves, is applicable to the manual test connection, vent and drain valves located at the given penetrations. GDC 57 requires manual containment isolation valves be closed and locked to maintain containment integrity. Locking the test connection, vent and drain valves represents a level of administrative control not previously considered to be required for these and other test connection, vent or drain valves.

"The general design basis governing containment building isolation requirements is leakage through all fluid penetrations, not serving accident-consequence limiting systems, must be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the containment building, and various types of isolation valves.

"Based on ANO's licensing basis and NRC regulations, no specific requirement to lock these valves previously existed. Industry guidance indicates that test connection, vent and drain valves should not be classified as GDC isolation valves, but as administratively con trolled containment system barriers. ANO has procedurally controlled these valves to ensure they are closed during power operations.

"The general design basis governing containment isolation requirements is given in Section 5.2.5 of the current Arkansas Nuclear One- Unit 1 (ANO-1) Safety Analysis Report. The ANO-1 Safety Analysis Report Table 5.1, Reactor Building Isolation Valves, lists GDC 55, 56, and 57 penetrations and the applicable containment isolation valves. There are no test connection, vent and drain valves identified in the table. The definition of ANSI/ANS-56.2-1984, Containment Isolation Provisions for Fluid Systems, clearly distinguish between containment isolation valves and the containment penetration test connection, vent and drain valves by defining test connections or vents as being provided so that containment isolation valves can be tested. Figures 1 and 2 of ANSI/ANS-56.2-1984 demonstrate the criteria for GDC 55, 56, and 57 and includes only the main process line valves (main feedwater block valves for P-3 and P-4), and do not demonstrate provision for test connection, vent or drain valves."

In Section 6.2.3 of the ANO-1 Safety Evaluation Report (SER), it states, in part, "the reactor building isolation system is designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided so that no single valve or piping failure can result loss of containment integrity." The SER further states that "the NRC staff has reviewed the containment isolation system for conformance to GDC 55, 56, and 57." The NRC concluded that, the containment isolation system meets the intent of the general design criteria. From the Final Safety Analysis Report (FSAR) description of the containment building isolation system it is evident that only the process line valves were included as containment building isolation valves. Test connection, vent and drain valves on these penetrations are neither depicted nor described."

The inspectors noted that the process valves that the licensee discussed in their response to the violation are some of the valves for which the licensee now has deleted GDC 57 applicability.

The inspectors also found during the review that, during the initial licensing process for Unit 1, the Commission Staff's Request for Additional Information 5.83 requested information regarding compliance with GDC 57 for the Unit 1 main feedwater lines to each steam generator and was specifically evaluated by the Commission's Staff to determine if the piping configuration used complied with GDC 57. In addition, Request for Additional Information 10.1 for Unit 1 requested information concerning the turbine stop valves serving as containment isolation valves for isolation of the unaffected steam generator in the event of a steamline rupture accident. The licensee responded to Question 10.1 that the main steam block valves, not the turbine stop valves, were designed to serve as the containment isolation valves and that these valves met Criterion 57 of 10 CFR Part 50, Appendix A. The staff also requested information regarding how the licensee was implementing GDC 57 for penetrations containing the steam generator sample valves for Unit 1.

Description of Change to the Unit 1 UFSAR

During review of this issue, the inspectors questioned the licensee concerning the licensing basis for the Unit 1 steam generator secondary side reactor building penetrations and the subsequent deletion of the GDC applicability for the following penetrations:

Containment Penetration P1	Main Steam from Steam Generator E-24A to EFW and Main Turbine
Containment Penetration P2	Main Steam from Steam Generator E-24B to EFW and Main Turbine
Containment Penetration P3	Main Feedwater to Steam Generator E24-A
Containment Penetration P4	Main Feedwater to Steam Generator E-24B
Containment Penetration P10	Steam Generator Sampling
Containment Penetration P17	EFW to Steam Generator E-24A
Containment Penetration P65	EFW to Steam Generator E-24B

Based on the inspectors' questions, the licensee performed a review of the Unit 1 design and licensing bases and developed a position paper, which is Attachment 2 to this inspection report. In the licensee’s position paper, the licensee stated "The containment boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes, the outer surface of the lines emanating from the steam generator, and the outer surface of the steam generator between the tube sheets. This position is based on the concept of treating the secondary system pressure boundary as an extension of the containment liner." The licensee further stated that, although this position was not well documented, this position has always been the understanding of plant personnel familiar with the original design and licensing basis. The licensee further stated that this concept was based on Westinghouse Document WCAP-7451, dated September 1972. The inspectors could not identify that this document describing the containment boundary was ever referenced in any licensing correspondence with the Commission that was applicable to Unit 1.

The licensee stated that, ultimately, the concept of the secondary pressure boundary being an extension of containment boundary was explicitly approved by the NRC in the Safety Evaluation Report approving renewal of the Unit 1 operating license. This was based on the NRC's Request for Additional Information 2.3.2.7-2.

The Request for Additional Information stated "In P&ID LRA-M-237, Sheet 1, the redundant isolation valves (SS-1017B and SS-1018B) for the test connections of the sampling system are not highlighted as being within the scope of license renewal. However, containment isolation provisions require double isolation at the test connections for greater assurance of containment integrity. Provide a justification as to why the second isolation valve on each test connection is not in-scope." The licensee's response, dated August 30, 2000, stated "Please see ANO-1 Safety Analysis Report, Table 5-1, Penetration 10, and note 8. This penetration is associated with the secondary side of the steam generator and is not subject to GDC 57. In accordance with the current licensing basis, the containment boundary or barrier against fission leakage to the environment is the inside surface of the steam generator tubes, the outer surface of the lines emanating from the steam generator, and the outer surface of the steam generator between the tube sheets. Therefore, Valves SS-1017B and SS-1018B do not meet the scoping criteria for license renewal."

The NRC found the licensee's response acceptable. However, the inspectors noted that the GDC 57 applicability for the secondary system penetrations had already been deleted without prior Commission approval with the administrative change that is the subject of this violation for the change to UFSAR, Table 5-1.

Based on the inspectors' review, the inspectors concluded that on March 31, 1995, the licensee performed an inadequate 10 CFR 50.59 screening and, as a result, the licensee had not performed a safety evaluation as required by 10 CFR 50.59.

Screening Review for Unit 1 changes

Approval for UFSAR, Amendment 14, was based on Engineering Report 93-R-0007-01, "Containment Penetration Design Summary Document," Revision 0. As stated in the previous section of this inspection report, the inspectors reviewed Engineering Report 93-R-0007-01 and the associated 10 CFR 50.59 screening review. The following note was included in the Basis for Change for all of the secondary system containment penetration valves:

"ADMINISTRATIVE/EDITORIAL - Information is confusing in that it leaves the impression that these valves are Reactor Building Isolation valves when they are not. Deletion is consistent with note 8" (Unit 1 UFSAR, Table 5-1).

Because the changes made for all of the secondary system containment penetration valves were viewed as administrative/editorial, these changes met the licensee's criteria for changes specifically exempted from further 10 CFR 50.59 evaluation. This decision was in accordance with Procedure 1000.131, Attachment 1, which was, at that time, the licensee’s procedure governing 10 CFR 50.59 determinations/evaluations.

The inspectors determined that the change in definition of containment was not an administrative change because changes in the potential for increased fission product release during a steam generator tube rupture accident or other postulated accidents were not evaluated.

(3) Analysis

This finding is not suitable for evaluation using the SDP. The finding is determined by management to be of very low safety significance. This finding is more than minor, because the change should have received NRC's review prior to implementation.

(4) Enforcement

The inspectors evaluated the change in the UFSAR documented description of containment and the change in the updated UFSAR list of containment isolation valves. At the time that the 10 CFR 50.59 screening was performed in 1995, the licensee did not perform a safety evaluation for the change in the UFSAR documented containment description or for the deletion of multiple valves from the UFSAR list of containment isolation valves and resultant deletion of Technical Specification applicability for these containment isolation valves. The failure to perform a safety evaluation was a violation of 10 CFR 50.59.

Under the previous 10 CFR 50.59 rule, an inadequate screening evaluation resulted in the licensee not identifying whether the change would result in an increase in the consequences of an accident previously evaluated or whether there was an increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report change should have received NRC approval prior to implementation, because the change constituted an unreviewed safety question.

To evaluate conformance with the current 10 CFR 50.59 rule, the inspectors reviewed the change in the UFSAR documented containment description, deletion of multiple valves from the UFSAR list of containment isolation valves, and the resultant deletion of the Technical Specification applicability for these containment isolation valves against the requirements of the new rule, using the guidance in Regulatory Guide 1.187 and NEI 96-07, Revision 1.

An evaluation under the new rule was not performed by the licensee. However, departure from GDC 57 does result in more than a minimal increase in the likelihood of occurrence of a malfunction of a component important to safety. Based on these findings, the inspectors concluded that a license amendment was also required by the current 10 CFR 50.59 rule.

Therefore, NRC will not exercise enforcement discretion pursuant to Section VII.B.6 of the Enforcement Policy for the 1995 failure to perform a safety evaluation as required by 10 CFR 50.59.

The violation is not suitable for SDP evaluation, but has been reviewed by NRC management and is determined to be a Green finding of very low significance. The licensee entered this finding into its corrective action program as Condition Report (CR) ANO-C-2003-0242. This violation is being treated as an example of a noncited violation (50-313/0302-01; 50-368/0302-01) consistent with Section VI.A of the NRC Enforcement Policy.

1R04 Equipment Alignment

.1 Partial System Walkdown (71111.04)

a. Inspection Scope

The inspectors performed a partial system walkdown of the Unit 2 service water system while maintenance was being performed on Pump 2P-4C on February 3-5, 2003. The inspectors verified proper component alignment and operation in accordance with Procedure 2104.029, "Service Water System Operations," Revision 53, and system piping and instrumentation diagrams to verify that the system was in a proper standby lineup. The inspectors also examined component material condition.

The inspectors performed a partial system walkdown of the Unit 1 service water system while maintenance was being performed on Pump P-4A on February 18-20, 2003. The inspectors verified proper component alignment and operation in accordance with Procedure 1104.029, "Service Water and Auxiliary Cooling System," Revision 55, and system piping and instrumentation diagrams to verify that the system was in a proper standby lineup. The inspectors also examined component material condition.

The inspectors performed a partial system walkdown of the Unit 2 vital dc electrical system during maintenance on Battery 2D-12 on March 14-15, 2003. The inspectors verified proper system alignment and operation in accordance with Procedure 2107.004, "DC Electrical System Operation," Revision 20, to verify that the system was in a proper standby lineup.

b. Findings

No findings of significance were identified

1R05 Fire Protection

.1 Routine Inspection (71111.05Q)

a. Inspection Scope

The inspectors referenced the Fire Hazards Analysis Report, Revision 7, during the following inspections to ensure that conditions were consistent with the requirements of the licensee's fire protection program for fire protection systems design, control of transient combustibles and ignition sources, fire detection and suppression capability, fire barriers, and any related compensatory measures. Additional documents reviewed included Procedure 2203.014, "Alternate Shutdown," Revision 14; Technical Guideline 2203.014, "Alternate Shutdown," Revision 14; "ANO Pre-Fire Plan," Revision 1; Procedure 1000.047, "Control of Combustibles," Revision 15; Procedure 2203.034, "Fire or Explosion," Revision 5; and CR ANO-2-2003-0229.

- Unit 2 Battery Room 2D11, Fire Zone 2103V, conducted on February 21, 2003
- Unit 1 Switchgear Room A3, Fire Zone 99M, conducted on February 19-21, 2003
- Unit 1 Cable Spreading Room, Fire Zone 97R, conducted on March 5-7, 2003
- Unit 1 Integrated Control System Relay Room, Fire Zone 97R on March 5-7, 2003
- Unit 1 North Battery Room, Fire Zone 95-O, on March 5-7, 2003
- Unit 2 Battery Room 2D12, Fire Zone 2102Y, conducted on March 11-13, 2003
- Unit 2 Core Protection Calculator Room, Fire Zone 2098C, conducted on February 10, 2003

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors evaluated and discussed with the licensee the risk assessments listed below to verify that assessments were performed when required and appropriate compensatory actions were taken. The inspectors reviewed these assessed risk configurations against actual plant conditions and any inprogress evolutions or external events to verify that the assessments were accurate, complete, and appropriate for the conditions. In addition, the inspectors walked down the control room and plant areas to verify that compensatory measures identified by the risk assessments were appropriately performed.

- Unit 1 Emergency Diesel Generator K4B risk assessment associated with the system outage conducted on January 17, 2003
- Unit 1 Emergency Diesel Generator K4A risk assessment associated with the system outage conducted on January 21-22, 2003
- Unit 2 Service Water Pump 2P-4C electrical power supply cable replacement conducted on February 3-4, 2003
- Unit 1 Service Water Pump P-4A electrical power supply cable replacement conducted on February 17-20, 2003
- Unit 2 Battery Cell 2D12 replacement and testing conducted January 10, 2003

- Unit 2 Battery Cell 2D12 replacement and testing conducted March 14-16, 2003

- Unit 1 Emergency Diesel Generator K4A maintenance activities conducted on February 14-19, 2003

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Nonroutine Plant Evolutions and Events (71111.14, 71153)

a. Inspection Scope

The inspectors observed the following nonroutine evolutions to verify that they were conducted in accordance with licensee procedures and Technical Specification requirements:

On February 2, 2003, Unit 2 experienced a failure of the operating Loop 2 Service Water Pump 2P-4C. Operators entered Abnormal Operating Procedure 2203.022, "Loss of Service Water," Revision 8, and promptly started Standby Service Water Pump 2P-4B to provide service water to Loop 2. The licensee initiated CR ANO-2-2003-0178 to document the event and corrective actions. As part of an extent of condition review, the licensee also initiated CR ANO-C-2003-0067 to review the condition of medium-voltage underground cables at the facility. From this review, the licensee identified that the cables for Unit 1 Service Water Pump P-4A should be replaced as a prudent measure. The cables for Pump P-4A were replaced during the week of February 17, 2003.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the correctness of evaluations, the use of compensatory measures, if needed, and compliance with the Technical Specifications. The inspectors' review included a verification that operability determinations were made as specified by the licensee's Procedure LI-102, "Corrective Action Process," Revision 2, and Procedure 1000.104, "Condition Reporting and Immediate Reportability Determinations," Revision 17. The technical adequacy of the determinations was reviewed and compared to the Technical Specifications, Technical Requirements Manual, UFSAR, and associated licensing-basis documentation, as appropriate. The operability determinations that were reviewed were documented in the following CRs:

- ANO-C-2003-00067 Service water pump power supply cable evaluation
- ANO-2-2002-00644 Installation of temporary alteration for installation of heaters to supplement local heating due to Battery Room 2D11 temperature below 65°F
- ANO-2-2003-00237 Various abnormal indications and alarms which occurred at roughly the same time on Unit 2
- ANO-2-2003-00383 Failed support rod from Hanger 2EBD-15-H2 which provides support from main steam header to stop Valve 2CV-0250
- ANO-1-2003-00225 EDG K-4A fuel oil sight glass leakdown problem
- ANO-1-2003-00193 Air entrained in the EDG 1 fuel oil return system
- ANO-2-2002-02084 Low voltage on Unit 2 Battery 2D12 Cell 40
- ANO-2-2003-00412 Low voltage on Unit 2 Battery 2D12 Cell 40
- ANO-2-2003-00193 Unit 2 low spent fuel pool temperature
- ANO-1-2003-00378 Emergency Feedwater Valve SV-2613 nameplate and documentation errors

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

.1 Annual Review (71111.17A)

a. Inspection Scope

The inspectors reviewed the plant modification described in Engineering Request (ER) ANO-2000-2688-002, "Uprate L-3 Spent Fuel Crane to Handle Capacity of 122 Tons." This modification replaced the previous Crane L-3 nonsingle-failure proof trolley, rated at 100 tons, with a new trolley of single-failure proof design rated at 130 tons. This modification was necessary to allow the licensee to lift spent fuel storage casks of a new, heavier design. The review included the safety evaluation prepared to determine if the modification required a license amendment per the requirements of 10 CFR 50.59.

b. Findings

The licensee's 10 CFR 50.59 evaluation concluded that the proposed modification did not require a license amendment. The licensee concluded that the upgraded crane design met the requirements of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Therefore, the licensee concluded that the upgraded crane design was acceptable for implementation without the need for an NRC license amendment.

The inspectors disagreed with this conclusion. The inspectors acknowledged the new crane was intended to meet single-failure-proof design standards and utilized a trolley design documented in a vendor topical report that was previously approved by the NRC. However, the inspectors noted that Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," identified that installation of a single-failure-proof crane design may reasonably be expected to eliminate most, perhaps 90 percent, of load drop probability, meaning the failure probability was not zero. The inspectors concluded that the increase in the maximum critical load rating of the crane (from 100-130 tons), combined with a required load path that would carry a loaded spent fuel storage cask over the control rooms, would require a license amendment.

The inspectors, managers from the NRC Region IV office, and representatives of the NRC Office of Nuclear Reactor Regulation (NRR) informed the licensee of this conclusion in a telephone call on February 13, 2003. The inspectors also informed the licensee that failure to submit a license amendment request for this modification was a potential violation of 10 CFR 50.59. The licensee entered this issue into its corrective action program as CR ANO-C-2003-0092. The licensee subsequently submitted a license amendment request for the Crane L-3 modification to the NRC on February 24, 2003. The license amendment request was still under review by the NRR staff at the conclusion of the inspection period. Although a potential violation of 10 CFR 50.59 was determined to exist, the inspectors considered this issue to be an unresolved item, pending completion of review of the license amendment request by the NRR staff and a determination of the significance of the violation (50-313/0302-02; 50-368/0302-02). The upgraded crane had not been used to transport a loaded spent fuel storage cask and was under administrative controls preventing its use in this manner, pending resolution of this issue. Therefore, this issue had no immediate safety significance.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

For the maintenance activities identified below, the inspectors observed the postmaintenance testing activities in the control room or locally and/or reviewed the test data obtained from the field. The inspectors observed whether the tests were performed in accordance with procedures, acceptance criteria were consistent with Technical Specifications, and the results recorded met the test acceptance criteria. In addition, the inspectors verified that any deficiencies were recorded and resolved. These activities included:

- Unit 1 Emergency Diesel Generator 1 testing in accordance with Procedure 1104.036, "Emergency diesel Generator Operation," Revision 41, conducted on January 24, 2003

- Unit 1 Emergency Diesel Generator 1 testing in accordance with Procedure 1104.036, "Emergency Diesel Generator Operation," Revision 41, conducted on February 19, 2003

- Unit 2 2D12 battery testing for Cell 40 replacement conducted on March 10-15, 2003, as described in the maintenance action plan associated with CR ANO-2-2003-0412

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed from either the control room or locally the performance of and/or reviewed the documentation for the following surveillance tests. This was done to verify that the surveillance tests were performed in accordance with approved licensee procedures and met Technical Specification requirements. In addition, the applicable test data was also reviewed to verify whether they met Technical Specifications, UFSAR, and licensee procedure requirements.

- Procedure 2104.036, "2DG1 Monthly Test (Slow Start)," Supplement 1B, Revision 46, performed on January 29, 2003

- Procedure 2105.009, "CEA Exercise Test," Supplement 2, Revision 21, performed on March 21, 2003

- Procedure 2403.023, "2D12 Quarterly Surveillance," Revision 14, performed on March 17, 2003

- Procedure 2403.023, "2D12 Quarterly Surveillance," Revision 14, performed on March 19, 2003

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

On January 28-30, 2003, the inspectors reviewed the implementation of Procedure 2106.032, "Unit 2 Freeze Protection Guide," Revision 9, Section 8, "Battery Room Low Temperature." This procedurally controlled temporary alteration installed a temporary portable 15 Kw electric heater in Unit 2 Corridor 2104 to provide supplemental heating for the safety-related battery rooms in severe wintertime conditions. The justification for the temporary non-Q battery room heating was evaluated and documented in ER ANO 2002-0145-000. The inspectors confirmed that this temporary alteration was properly installed as authorized by the procedure and the ER.

On February 3-4, 2003, the inspectors reviewed the implementation of Maintenance Action Item (MAI) 79434, which installed a temporary alteration evaluated by ER ANO-2003-0098-00. This temporary alteration installed a temporary power supply to the Unit 2 Service Water Pump 2P-4C motor while its permanent power supply cables were being replaced. The inspectors confirmed that this temporary alteration was implemented and installed as authorized by MAI 79434 and Procedure 1000.028, "Control of Temporary Alterations," Revision 25.

On February 19-21, 2003, the inspectors observed that Fire Door 48 had been removed in support of the power supply cable replacement for the Unit 1 Service Water Pump P-4A motor. The door had been removed per the instructions of MAI 80036. During the week of March 3, 2003, the inspectors reviewed this maintenance activity to determine whether an unevaluated temporary alteration to the facility had been performed.

b. <u>Findings</u>

A Green noncited violation was identified for performing a maintenance activity that resulted in the implementation of an unevaluated temporary alteration. Removal of Door 48 was not evaluated for its impact on safety-related equipment following a potential high energy line break or for its impact on room cooling capability performance.

Door 48 is a 3-hour rated fire door separating safety-related switchgear Room A3 from the turbine building. Located outside switchgear Room A3 are two high-pressure feedwater heaters. The inspectors reviewed the Unit 1 UFSAR, "An Evaluation of Pipe Breaks Outside the Reactor Building," Section A.7, Amendment 17, and noted that the main feedwater pipe break analysis stated that steam and water from a postulated feedwater pipe break are prevented by walls from entering into the switchgear rooms. The inspectors concluded that this statement implied that doors in the subject walls were also credited as barriers. The inspectors also reviewed Procedure 1104.027, "Battery and Switchgear Emergency Cooling System," Revision 20. This procedure identified that the emergency cooling system was designed to limit maximum room temperature to 104°F in switchgear Room A3 with turbine building ambient air temperatures at 122°F. The inspectors also reviewed the component database for Door 48 and found that this door was not identified as having a design function as a high-energy line break barrier or room temperature maintenance barrier.

Door 48 was removed per MAI 80036 on February 17, 2003, in order to support cable replacement activities for Service Water Pump P-4A and was returned to service on February 20, 2003. The MAI did not include an engineering evaluation for the acceptability of removal of Door 48 nor was there one performed.

The inspector concluded that removal of Door 48 resulted in a change to the design function of the barrier separating switchgear Room A3 and the turbine building and was a temporary alteration. Procedure 1000.028, "Control of Temporary Alterations," Revision 23, requires that all temporary alterations receive an engineering review documented in an ER.

The licensee entered this issue into its corrective action program as CR ANO-1-2003-0258. Per UFSAR, Section A.7.3, a postulated main feedwater pipe break outside of Door 48 was a credible event. With Door 48 removed, the resultant harsh environment from a break would have potentially incapacitated one of the redundant trains of equipment necessary to achieve safe shutdown conditions. However, the remaining train of equipment would have remained unaffected. The condition did not result in equipment being inoperable because equipment in switchgear Room A3 was capable of performing its intended function during a design basis event (e.g., a loss-of-coolant accident) during the time Door 48 was removed.

As part of its corrective action for this finding, the licensee surveyed all doors that separate the turbine building from safety-related equipment areas. All doors that were discovered to also have a design function to provide a high energy line break barrier were identified and were provided administrative controls requiring an engineering evaluation prior to their removal.

Unit 1 Technical Specification 5.4.1.a requires that written procedures shall be established, implemented, and maintained, covering activities identified in Regulatory Guide 1.33, Revision 2, Appendix A, which includes modification work. Procedure 1000.028 required an engineering evaluation be performed to support a temporary alteration. Failure to perform an engineering evaluation for removal of Door 48 was a violation. The safety significance of removal of Door 48 was determined to be very low since this issue screened as Green during a Phase 1 SDP evaluation. The issue was considered more than minor because it affected the mitigating systems cornerstone objective for design control and modifications, because the ability to mitigate the consequences of a high energy line break would have been affected if the finding had affected more than one train of equipment. This violation is being treated as noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-313/0302-03).

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed an in-office review of Revision 27 to the Arkansas Nuclear One Emergency Response Plan, received January 25, 2002, and Procedure 1903.010, "Emergency Action Level Classification," Revision 37. The inspector compared the current revisions with their previous revisions and 10 CFR 50.54(q) to determine if the current revision decreased the effectiveness of the emergency plan.

b. Findings

No findings of significance were identified.

2. Radiation Safety
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors reviewed radiation work permits (RWPs) from Refueling Outages 2R15 and 1R17 that both accrued more than 5 rem and that had actual doses that exceeded the initial work activity dose estimates by more than 50 percent. No high exposure jobs or work activities in high radiation areas were performed during the inspection; however, the inspectors observed the movement of Cooling Coil VUC1B (in accordance with RWP 2003-1040) and the replacement of Spent Fuel Filter F4 (in accordance with RWP 2003-1017). During the work activities, the inspectors made independent radiation measurements and observed ALARA practices and contamination control practices. The following specific items were reviewed and compared with regulatory requirements:

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- ALARA program procedures
- RWP 2002-1420, "Scaffolding and Insulation Activities During 1R17"; RWP 2002-1452 and RWP 2002-1515, "Reactor Head CRD Nozzle Inspection"; RWP 2002-2442, "Steam Generator Inspection and Maintenance"; and RWP 2002-2507, "Pressurizer Heater Repair"
- Radiological work planning
- Use of engineering controls to achieve dose reductions
- Processes used to estimate and track exposures

- Selected corrective action documentation involving higher than planned exposure levels and radiation worker practice deficiencies since the last inspection in this area (CRs ANO-C-2001-00297, ANO-2-2002-00882, ANO-C-2003-00034, and ANO-C-2003-00118)

b. Findings

Introduction: The inspectors identified a Green noncited violation because the licensee failed to provide an adequate reason for revising RWPs, as required by a Technical Specification required procedure.

Description: During a review of RWP packages, the inspectors noted multiple examples of the licensee adjusting work activity dose estimates without providing adequate bases. The dose estimates were adjusted to account for failures to control work activity doses and failed to implement additional means to control dose.

For example, RWP 2002-1420, "Scaffolding and Insulation Activities During 1R17," Revision 4, increased the work activity dose estimate from 15.578-16.578 rem. The documented reason stated, "The dose estimate would have been exceeded in the next shift. Additional dose was added to complete work." The licensee provided no other justification for the work activity dose estimate adjustment. RWP 2002-1452, Revision 2, increased the work activity dose estimate from 4.2-7.9 rem. The documented reason stated, "Made corrections to EAD setpoints in the special instruction to match the ERIMS trigger on the RWP." While this was a legitimate reason for a RWP revision, it did not address the basis for the adjustment of the work activity dose estimate. In the most significant example with regard to actual dose exceeding the legitimate work activity estimate, RWP 2002-2507, Revision 2, increased the work activity dose estimate from 4.6-6.893 rem. The reasons cited for the revision were: (1) vendor equipment did not machine correctly, (2) workers were less efficient because of heat stress conditions, and (3) mockup facilities used for training were inconsistent with actual plant components. RWP 2002-2507, Revision 3, increased the work activity dose estimate from 6.893-7.0 rem. The reasons cited for the revision were: (1) the current projection was insufficient to cover the work on two remaining mechanical nozzle sealing assembly clamps and (2) additional management was in the field to provide oversight for inexperienced workers.

With respect to RWP 2002-2507, the inspectors found that equipment ineffectiveness was a legitimate reason in some cases for revising dose estimates. However, the licensee had previously cited equipment ineffectiveness as a basis for Revision 1 of the RWP, in which the work activity dose estimate was increased from 3.660-4.600 rem. No additional justification was provided to support the use of equipment ineffectiveness a second time to justify an increase to the dose estimate. Licensee representatives stated that similar environmental conditions existed in previous refueling outages. Therefore, containment environmental conditions should have been anticipated. The inconsistent mockup facility resulted from plant drawing discrepancies and, according to licensee personnel, was not a recent development. Therefore, the licensee had sufficient opportunity to identify and correct the situation before the refueling outage. The inspectors determined through interviews with licensee personnel that the work on all mechanical nozzle sealing assembly clamps was considered in the original dose estimate and the number of clamps did not increase. The use of inexperienced workers did not begin at this stage of the work and should have been considered from the start and addressed through additional training. The inspectors, after consultation with NRR, concluded that the bases for the work activity dose estimate adjustments in RWP 2002-2507, Revisions 2 and 3, were invalid because there were inadequate justifications for the adjustments. Consequently, the actual dose for this work activity (RWP 2002-2507) exceeded 5 rem (7.165) and exceeded the legitimate dose estimate (4.6 rem) by more than 50 percent.

Analysis: The inspectors determined that this finding was associated with the Occupational Radiation Safety Cornerstone program and process attributes (ALARA planning/projected dose) and affected the objective of the cornerstone, which is to protect the worker from exposure to radiation. Therefore, the finding was greater than minor. The occurrence involved a failure to maintain or implement, to the extent practical, procedures needed to achieve occupational doses that were ALARA, which resulted in unplanned, unintended occupational collective dose for a work activity. Therefore, the safety significance of the finding was evaluated using the Occupational Radiation Safety SDP. However, because the licensee's 3-year rolling average collective dose was not greater than 135 person-rem/unit, the finding had no more than very low safety significance.

Enforcement: Unit 1 Technical Specification 5.4.1.a and Unit 2 Technical Specification 6.8.1.a require that procedures be established, implemented, and maintained covering the applicable procedures in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 7.e.1, includes procedures for an RWP program. Procedure NMM RP-105, "Radiation Work Permits," Revision 1, implements this requirement. Procedure NMM RP-105, Section 5.8.3, requires that the reason for a revision be recorded on the RWP Revision/Edit form. However, the licensee failed to record the reason for the revision of RWPs when it adjusted work activity dose estimates without providing an adequate bases. Because the failure to record the reasons for RWP revisions was of very low safety significance and has been entered into the corrective action program as CR ANO-C-2003-0031, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-313/0302-04; 50-368/0302-04).

4. Other Activities [OA]

4OA2 Identification and Resolution of Problems

.1 Cross-Reference to Problem Identification and Resolution Findings Documented Elsewhere

Section 2OS2 describes a problem involving the lack of justification for adjusting work activity dose estimates. The NRC identified a previous example of this problem in May 2001. The problem resulted in a noncited violation, which was documented in NRC Inspection Report 50-313/2001-02; 50-368/2001-02. The licensee initiated

CR ANO-C-2001-00297 to document the problem and track the corrective actions. This CR was closed in February 2002 after the corrective actions were implemented. The corrective actions proved ineffective to prevent the recurrence of the problem.

4OA3 Event Followup (71153)

.1 (Open) Licensee Event Report (LER) 50-368/2002-002-00: Automatic actuation of the reactor protection system caused by a main turbine trip due to failure of the main generator reverse power relay resulted in a reactor trip.

This LER was issued during this inspection period for an event that occurred on December 19, 2002. Continued evaluation of the root cause and significance will be performed and documented in a future inspection report.

.2 (Closed) LER 50-368/2001-001-00: Crediting a designated operator for manual action during surveillance tests affected operability of the emergency feedwater system if condensate pumps had been lost during the tests.

In May 2001, an NRC inspection identified that the licensee's method of surveillance testing of the Unit 2 emergency feedwater system resulted in inappropriately crediting operator actions and resulted in operation prohibited by Technical Specifications. The method of testing employed using the startup and blowdown demineralizer system as a source of emergency feedwater suction. If a loss of condensate pumps occurred, both trains of the emergency feedwater system were potentially incapable of performing their intended function. The NRC documented the finding, enforcement, and safety significance in NRC Inspection Report 50-313/2001-02; 50-368/2001-02. The licensee initiated CR ANO-2001-0349 to address the issue and corrective actions. The licensee reported this condition of using the startup and blowdown demineralizer system as a source of emergency feedwater pump suction and inappropriately crediting manual actions during surveillance testing in the LER.

The inspectors reviewed the LER and concluded that the licensee's root cause determination was adequate and the completed and proposed corrective actions were appropriate. This LER is closed.

.3 (Closed) LER 50-313/2001-003-00: Control room emergency ventilation system radiation monitors were inoperable due to an inadequate procedure change caused by training and system knowledge deficiencies.

On April 17, 2001, the licensee identified that maintenance was performed on auxiliary building ventilation supply fans while the normal control room ventilation dampers were closed and neither of the control room emergency ventilation supply fans was running. The licensee determined that both trains of control room emergency ventilation system radiation monitors were inoperable. With the normal ventilation dampers closed, the ability of these detectors to generate a radiation signal for control room emergency ventilation initiation would be significantly degraded without air flow representative of the activity entering the control room. This was a violation of Unit 1 Technical Specification, Table 3.5.1-1, and Unit 2 Technical Specification, Table 3.3-6. This was also reported as a condition that could have prevented the fulfillment of the safety function of a system that is needed to mitigate the consequences of an accident.

The licensee initiated CR ANO-2001-0207 to address the issue and corrective actions. The licensee reported this condition and its inadequate procedural guidance for radiation monitor operability in the LER.

The inspectors reviewed the LER and concluded that the licensee's root cause determination was adequate and the completed and proposed corrective actions were appropriate. This finding was considered to be of very low safety significance because the condition existed less than 8 hours and another control room area radiation monitor was available to automatically initiate emergency ventilation if a high radiation condition in the control room existed. This violation of NRC requirements meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a noncited violation.

4OA5 Other Activities

.1 The inspectors reviewed information only the Accreditation Evaluation Report, issued by the Institute of Nuclear Power Operations National Academy of Nuclear Training Nuclear Training Accrediting Board of ANO training programs, performed in May 2002.

4OA6 Meetings, Including Exit

The inspectors presented the results of the ALARA Planning and Controls inspection to Mr. Craig Anderson, Vice President, Operations, and other members of the licensee's staff at the conclusion of the inspection on January 31, 2003. The licensee acknowledged the findings presented. The inspectors conducted a followup telephonic exit meeting with Mr. Glenn Ashley, Licensing Manager, and other members of the licensee's staff on February 20, 2003.

The inspectors presented the inspection results of the emergency plan change review to Mr. R. Fowler, Acting Supervisor, Emergency Planning, during a telephonic exit interview conducted on February 20, 2003. The licensee acknowledged the findings presented.

The resident inspectors presented the inspection results of the resident inspections to Mr. Craig Anderson, Vice President, Operations, and other members of the licensee's staff on April 3, 2003. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

40A7 Licensee-Identified Findings

The following findings of very low significance (Green) were identified by the licensee. They are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a noncited violation.

.1 On March 19, 2003, the licensee identified that Unit 1 turbine-driven emergency feedwater pump steam admission bypass Valves SV-2613 and SV-2663 were not environmentally qualified in accordance with 10 CFR 50.49. The safety significance of this finding was very low because the valves remained operable. This issue was entered into the licensee's corrective action program as CR ANO-1-2003-0369.

.2 On April 17, 2001, the licensee identified that the radiation monitors for both trains of the control room emergency ventilation system were inoperable. This event is discussed in Section 4OA3.

ATTACHMENT 1
SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

C. Anderson, Vice President, Operations
B. Beaird, Supervisor, Systems Engineering
S. Bennett, Licensing
M. Byram, Senior Lead Engineer
M. Chisum, Manager, Systems Engineering
M. Cooper, Licensing Specialist
S. Cotton, Director, Nuclear Safety Assurance
G. Dobbs, Design Engineer
B. Eichenberger, Manager, Unit 1 Operations
C. Eubanks, General Manager, Plant Operations
D. Fouts, Supervisor, Nuclear Engineering
R. Fowler, Acting Supervisor, Emergency Planning
M. Harris, Manager, Dry Fuel Storage Project
D. Hawkins, Licensing Specialist
A. Heflin, Acting Manager, Planning, Scheduling and Outages
J. Hoffpauir, Plant Manager, Operations
D. James, Manager, Engineering Programs and Components
J. Kowalewski, Director, Engineering
T. Mitchell, Manager, Unit 2 Operations
T. Nickels, Superintendent, Radiation Protection
B. Patrick, Operations Supervisor, Radiation Protection
D. Phillips, Design Supervisor
D. Scheide, Licensing Specialist
L. Schwartz, Manager, Nuclear Engineering
J. Sigle, Shift Assistant Operations Manager
D. Stoltz, Supervisor, Radiation Protection
R. To, Design Engineer
C. Tyrone, Manager, Quality Assurance
D. Williams, Senior Staff Engineer, Nuclear Engineering

C. Zimmerman, Plant Manager, Support

ITEMS OPENED

50-313/0302-02; 50-368/0302-02	URI	Failure to obtain a license amendment for upgrade of the spent fuel area crane (Section 1R17)

ITEMS OPENED AND CLOSED

50-313/0302-01; 50-368/0302-01	NCV	Deletion of containment integrity controls for secondary system containment penetrations (Section 1R02)
50-313/0302-03	NCV	Unauthorized temporary alteration (Section 1R23)
50-313/0302-04; 50-368/0302-04	NCV	Failure to provide adequate justifications for work activity dose estimate adjustments (Section 2OS2)

ITEMS CLOSED

50-368/2001-001-00	LER	Crediting a designated operator for manual action during surveillance tests affected operability of the emergency feedwater system if condensate pumps had been lost during the tests (Section 4OA3)
50-313/2001-003-00	LER	Control room emergency ventilation system radiation monitors were inoperable due to an inadequate procedure change caused by training and system knowledge deficiencies (Section 4OA3)

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
CIAS	containment isolation actuation signal
CR	condition report
ER	engineering request
FSAR	Final Safety Analysis Report
GDC	general design criterion

LCO	limiting condition for operation
LER	licensee event report
MAI	maintenance action item
NRR	Office of Nuclear Reactor Regulation
RWP	radiation work permit
SDP	significance determination process
SER	safety evaluation report
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 2

ANO PAPER

"Secondary System Pressure Boundary as an Extension of Containment Liner"