ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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License Nos.:	DPR-51
Report No.:	50-313/96-25 72-13/96-02
Licensee:	Entergy Operations, Inc.
Facility:	ANO, Unit 1
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	December 3-18, 1996
Inspectors:	J. V. Everett, Health Physicist J. F. Melfi, Resident Inspector T. H. Andrews, Radiation Specialist T. J. Kobetz, Project Manager A. D. Gaines, Health Physicist
Approved By:	D. Blair Spitzberg, Ph.D., Chief Nuclear Materials Licensing Branch
No Inspection:	No inspection of Unit 2 was performed.
Attachment:	Supplemental Information

EXECUTIVE SUMMARY

ANO, Units 1 and 2
NRC Inspection Report 50-313/96-25; 72-13/96-02

This inspection included direct observation by the inspectors of the movement of ANO Unit 1 spent fuel from the spent fuel pool to the ISFSI located within the protected area of the ANO site. All phases of the fuel movement activities were observed starting with the initial spent fuel movements in the spent fuel pool and concluding with the placement of the concrete storage cask on the ISFSI pad.

- The Technical Specifications in the VSC-24 Certificate of Compliance had been incorporated into the ANO procedures and were complied with during the process of loading, moving, and storage of the first VSC-24 cask at ANO (Section 1).

- The ANO health physics organization implemented a strong health physics program to control radiation exposure and contamination during all cask loading and moving operations. All ANO personnel involved with the dry cask storage program were observed to fully cooperate with the implemented radiological controls and demonstrated individual commitment to complying with the radiation work permit requirements for the project (Section 2).

- Difficulties were experienced during the early phases of fuel loading when the licensee inserted the fuel assemblies into the sleeves in the basket. These difficulties, in part, resulted from not having the basket sleeves aligned with the spent fuel racks, by the use of an extremely conservative underload setting on the spent fuel crane for the first few inches of the insertion process, and by the unexpected clouding of the water around the top of the basket. ANO identified and implemented adequate corrective actions for each of the problems (Section 3).

- Hydrogen monitoring during the welding of the shield lid was effectively implemented by ANO. On one occasion, a measurable level of hydrogen was observed for a short period of time. The maximum hydrogen level reached during welding activities was 7 percent of the Lower Explosive Limit for a couple of seconds. The continuous flow of air under the shield lid during the welding effort was effective in preventing hydrogen build-up and potential explosion problems (Section 4).

- The Safety Analysis Report for the VSC-24 cask included design specifications and detailed drawings. Any deviation from these design specifications, whether the deviation was a one-time nonconformance that affected an individual cask or a permanent design change that affected several casks, required that ANO have conducted an evaluation as specified in 10 CFR 72.48. Several nonconformances were accepted for the ANO transfer cask without completing a 10 CFR 72.48 safety evaluation. This was identified by the inspectors as a Severity Level IV violation (Section 5).

Report Details

Summary of Plant Status

Arkansas Nuclear One had developed and implemented a dry cask storage program using the Sierra Nuclear VSC-24 cask. Fourteen casks had been constructed at ANO for the loading of both Unit 1 and Unit 2 fuel. Additional casks will be constructed in the near future. In May 1996, ANO conducted a dry run exercise of their cask loading procedures. An NRC inspection team was present during the week of the dry run to conduct a preoperational inspection. The preoperational inspection identified no issues that would have prevented the successful loading of the ANO casks.

The actual loading of the fuel into the VSC-24 casks at ANO was planned for June 1996 but was delayed due to the hydrogen incident at Point Beach involving a VSC-24 cask. The NRC issued ANO a Confirmatory Action Letter (CAL) on June 3, 1996, and a supplement to the CAL on June 21, 1996, that restricted ANO from loading fuel or placing the VSC-24 cask into the spent fuel pool until the issues with the Point Beach hydrogen problem had been resolved. On December 3, 1996, the NRC issued a letter to ANO lifting the restrictions of the CAL and Supplement and allowed ANO to initiate fuel loading into the VSC-24 casks. Arkansas Nuclear One immediately started the preparations to move fuel from the Unit 1 spent fuel pool into a VSC-24 cask.

The VSC-24 design consisted of three main components: the multi-assembly sealed basket (MSB), multi-assembly transfer cask (MTC), and ventilated concrete storage cask (VCC). These three main components comprised the Sierra Nuclear Ventilated Storage Cask (VSC) system. The basket was designed to hold the fuel assemblies and had an internal structure of 24 sleeves, each sleeve capable of holding one fuel assembly. The transfer cask provided shielding during fuel loading and movement. The basket remained in the transfer cask throughout the phases of loading, welding, leak testing, and transfer to the concrete cask. The basket was transferred into the concrete storage cask in the train bay by placing the transfer cask on top of the storage cask, opening the doors on the bottom of the transfer cask, and lowering the basket into the storage cask. The transfer cask was then removed and placed in a temporary storage location on the turbine deck. The basket and storage cask were transported to the ISFSI pad on a train car and moved to an assigned location on the ISFSI concrete pad for long-term storage.

The first fuel assembly was placed into the multi-assembly sealed basket on December 4, 1996, at 9:30 p.m. Loading of all 24 fuel assemblies into the basket was completed December 6, 1996, at 5:43 a.m. Loading of the basket took approximately 32 hours. The shield lid and structural lid were welded onto the basket, and the fuel was moved to the ISFSI on December 18, 1996, at approximately 5 p.m.

The ANO organization responsible for the dry cask storage program demonstrated a high priority to safety and to completing procedural tasks correctly and completely. Strong management controls and oversight were demonstrated throughout the process. Each issue and problem was thoroughly evaluated. The effect of making a change in the procedures was always evaluated from the perspective of the total effect on safety of the project. Safety concerns were continuously reinforced by the management team over the need to meet the schedule. Management coordination and control, through the use of an operations center manned 24 hours per day, demonstrated excellent planning. Problems that occurred were quickly brought to the attention of management, and technically sound decisions were made throughout the length of the project.

The cask loading and supporting procedures were inclusive and provided good detail for completion of tasks. As the project continued, improvements in the process were identified for incorporation into future revisions of the procedures. As improvements were identified for activities in progress, personnel were available in the operations center 24 hours per day to review the requested changes and implement immediate procedural revisions. This responsive process resulted in completing a number of tasks efficiently.

1 VSC-24 Certificate of Compliance Technical Specifications (60855)

1.1 Inspection Scope

The Sierra Nuclear VSC-24 Certificate of Compliance No. 1007 included Technical Specifications and requirements for the use of the VSC-24 dry cask storage system. Arkansas Nuclear One had incorporated these Technical Specifications into their dry cask storage program procedures. This inspection included a review of ANO's compliance with the applicable Technical Specifications for the VSC-24 design.

1.2 Observations and Findings

a. Fuel Specification

Section 1.2.1 and Table 1 of the Certificate of Compliance established the characteristics of the spent fuel allowed for storage in the VSC-24 cask. The fuel must have a post irradiation time of $\geq$ 5 years, a decay power limit per fuel assembly of $\leq$ 1 kilowatt (kW) (i.e., 24 kW limit for the total cask), a maximum enrichment of $\leq$ 4.2 percent uranium-235 (U-235), and is limited by a maximum and minimum burnup, neutron flux, and gamma flux specified in the Safety Analysis Report. The fuel must also have no known gross cladding failures.

Arkansas Nuclear One Unit 1 fuel is Babcock & Wilcock, Mark B, 15 x 15 zircaloy clad fuel. Prior to commencing fuel loading of the basket, the licensee inspected candidate fuel assemblies using an underwater periscope. Seventy-two fuel assemblies were identified as acceptable. Of these, 48 assemblies were placed in a special location in the spent fuel pool and segregated into primary and alternate candidates for loading into the basket. For the 24 fuel assemblies selected for the first basket, each fuel assembly was assigned a specific location in the basket. Placement of a fuel assembly in a location other than the assigned location would have required approval of both reactor engineering and operations. For this first basket, all fuel elements were placed in originally assigned locations.

The fuel decay power per fuel assembly was limited by the Certificate of Compliance to 1 kW. The ANO fuel assemblies ranged from 0.19 kW to 0.26 kW. The total decay power for the 24 fuel assemblies placed in the first basket was approximately 5.2 kW. This fuel decay power is well below the limit of 24 kW total established in Table 1 of the Certificate of Compliance.

The maximum fuel burnup allowed was 51,800 megawatt day/metric tons uranium (MWd/MTU). The minimum burnup allowed was determined using figure 6.1-1 of the Safety Analysis Report. Based on this figure, the minimum burnup was 15,217 MWd/MTU. The fuel loaded into the first cask ranged from 15,780 MWd/MTU to 19,905 MWd/MTU.

The minimum post irradiation time authorized for the fuel by the Certificate of Compliance was 5 years. The fuel placed in the first cask ranged in age from 17.3 years to 19.5 years.

The maximum initial enrichment established in Table 1 is ≤ 4.2 weight percent U-235. The fuel placed in the first basket had a maximum enrichment of 2.067 weight percent U-235. The maximum weight of the fuel assemblies established in Table 1 is 1516 lbs. The ANO fuel assemblies weighed 1514 lbs.

The maximum gamma flux limit for each fuel assembly is 6.8×10^{15} photons/second with the spectrum bounded by Table 5.2-1 of the Safety Analysis Report. For neutrons, the limit for each fuel assembly is 1.2×10^{8} neutrons/second with the spectrum bounded by Table 5.2-2 of the Safety Analysis Report. The ANO fuel placed in the first cask was bounded by the spectrums in Tables 5.2-1 and 5.2-2. The maximum gamma flux noted was 3.56×10^{15} photons/sec. The maximum neutron flux noted was 5.21×10^{7} neutrons/sec.

All required parameters for the fuel as established by the Certificate of Compliance, Table 1, were met by the fuel placed in the first cask at ANO.

b. Maximum Permissible MSB (basket) Leak Rate

Section 1.2.2 of the Certificate of Compliance established the requirements for the helium leak testing of the basket. No leak greater than 1.0×10^{-4} standard cubic centimeters per second (scc/sec) at 0.5 atmospheres differential pressure would be allowed. Any leak observed greater than this authorized leak rate must be repaired.

Arkansas Nuclear One Procedure 1415.048, "Helium Leak Detection for Dry Fuel Storage MSB," Revision 1, was used by the licensee to perform the helium leak test. Procedure 1415.048 was developed based on guidance from ANSI N14.5 American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment. The helium detector used for the examination was capable of detecting helium leaks as low as 4.0×10^{-5} scc/sec. The acceptance criteria in the Certificate of Compliance was 1.0×10^{-4} scc/sec. Prior to and immediately after performing the leak test, the helium detector was verified operable using a helium standard.

To perform the helium leak test on the basket lid welds, the basket was filled with helium to 1.5 atmospheres and a hand held helium sniffer was slowly moved along the welded surface to detect leaks. During the examination of the final weld on the shield lid, the helium leak detector reading exceeded the 1.0×10^{-4} scc/sec limit. The leak test procedure was repeated several times and a high reading was again noted in the same location, indicating a leak was present. A dye penetration test and visual inspection of the weld identified a crack running through the weld approximately 2-3 inches long. This crack was determined to also extend into the basket wall. No unacceptable indications had been found during the dye penetrant test of the root pass weld. The affected area was ground out to remove the crack and then rewelded. A subsequent dye penetrant test and helium leak test indicated the weld had been repaired. In addition, a hydrostatic test was performed and also indicated no leakage. The licensee determined that the most likely cause of the defect was lamellar tearing in the basket wall caused by stresses encountered during welding. No problems were encountered during the welding of the structural lid.

c. Maximum Permissible Air Outlet Temperature

Technical Specification 1.2.3 established a maximum air temperature limit at the outlet of the loaded storage cask (VCC) of 110 degrees Fahrenheit above ambient temperature. This limit was based on a cask loaded with 24 kW of fuel. For a lower kW fuel, the temperature limit was determined by calculation.

Arkansas Nuclear One used the formula in Section 4.4.1.1 of the Safety Analysis Report and calculated a maximum temperature differential for various cask heat loads (kW). On December 18, 1996, actual measurements were completed of the temperatures for the first storage cask. The ambient temperature at the time of the measurement was 32 degrees Fahrenheit. The temperatures measured at the outlets of the storage cask were 48 degrees Fahrenheit at one of the four vents and 59 degrees Fahrenheit at the other three vents. The maximum difference between the ambient temperature and the outlet vent readings was 27 degrees Fahrenheit. The calculated maximum temperature differential allowed by Section 4.4.1.1 of the Safety Analysis Report was 34 degrees Fahrenheit. The actual measured values were determined to be within the calculated maximum limit.

d. Maximum External Surface Dose Rates

Technical Specification 1.2.4 established maximum external dose rates for the storage cask. The combined gamma and neutron dose rates are limited to 20 millirem/hour (mrem/hr) on the sides, 50 mrem/hr on the top, and 50 mrem/hr at the air inlets and outlets.

The dose rates measured on the outside of the transfer cask during the time the transfer cask and basket were suspended over the cask loading pit and the basket was empty of water indicated gamma dose rates of 10 milliRoentgen/hour (mR/hr) and neutron dose rates of less than 1 mrem/hr. Since the transfer cask provided less shielding than the storage cask, the expected dose rates when the basket was placed in the storage cask were expected to be very low. Actual measurements of the storage cask radiation levels were conducted after the basket was lowered from the transfer cask into the storage cask in the train bay. The highest dose rate reading on the sides of the storage cask was 1.6 mrem/hr, the top of the storage cask was 1.9 mrem/hr, and at the air inlets and outlets was 5.1 mrem/hr. All dose rate readings were determined to be within the requirements of the Technical Specification.

e. Maximum MSB (basket) Removable Surface Contamination

Technical Specification 1.2.5 established limits on the amount of contamination that can remain on the external surface of the basket. The maximum beta-gamma contamination limit is 10^{-4} μCi/cm^2. This equates to 22,000 disintegrations per minute(dpm) per 100 cm^2. The maximum alpha contamination limit is 10^{-5} μCi/cm^2 (2,200 dpm/100 cm^2). These limits were established in the Certificate of Compliance to minimize the potential for exposures offsite from loose contamination and are based on the basket being covered over its entire surface with a known high level of contamination above these limits.

The gap between the basket and the transfer cask was smeared by ANO health physics personnel. The highest contamination levels measured were 12,000 dpm/100 cm^2 beta-gamma and less than 20 dpm/100 cm^2 alpha. After the basket had been lowered into the storage cask, a smear of the inside of the transfer cask was completed. The highest contamination levels measured on the sides of the transfer cask were 6,000 dpm/100 cm^2. On the cask doors on the bottom of the transfer cask, four samples were taken. These samples indicated beta-gamma contamination levels of 4,000 dpm/100 cm^2, 5,000 dpm/100 cm^2, 16,000 dpm/100 cm^2, and 28,000 dpm/100 cm^2. The average beta-gamma contamination level was below the 22,000 dpm/100 cm^2 limit. Alpha samples were all less than 20 dpm/100 cm^2. These contamination surveys indicated that the contamination levels on the basket were below the Technical Specification limits.

f. Boron Concentration in the MSB (basket) Cavity Water

Technical Specification 1.2.6 established a minimum boron concentration of 2850 parts per million (ppm) for any water present in the basket. The Technical Specification required that within 4 hours prior to placement of the first assembly into the basket that the boron level be confirmed. Two independent samples are to be taken and confirmed by two separate individuals.

The first fuel assembly was loaded into the basket on December 4, 1996, at 9:30 p.m. Boron samples had been taken at 7:21 p.m. and 7:28 p.m. indicating 3049 ppm boron and 3051 ppm boron, respectively. Each sample was analyzed by a different person. Analysis was conducted in accordance with ANO's Procedure 1605.005, "Determination of Boron," Revision 5, and recorded on Form 1052.002AR.

Technical Specification 1.2.6 also required that the boron levels be reconfirmed at intervals not to exceed 48 hours. The licensee reconfirmed boron levels in the basket on an ongoing basis every 3 to 6 hours. Boron levels ranged from 2997 ppm to 3058 ppm during the time the water was in the cask. No significant depletion of the boron levels over time was evident, and boron levels were maintained above the minimum required levels throughout the time the water was in the basket.

g. MSB (basket) Vacuum Pressure During Drying

Technical Specification 1.2.7 established the requirement for the vacuum drying of the basket. The basket must be vacuum dried to ≤ 3 millimeters of mercury (mm Hg) and held for 30 minutes. Then the basket is pressurized with helium to 22.1 pounds per square inch (absolute) (psia) ± 0.5 psia and leak tested. Upon completion of the leak test, a second vacuum drying was required which must also meet the criteria of ≤ 3 mm Hg for 30 minutes.

Arkansas Nuclear One started the vacuum drying process on December 10, 1996, at 11 a.m. The water had been drained from the basket by backfilling it with helium. A vacuum system which included a pressure gauge connected to the top of the basket was then connected. After the first day of vacuum drying, the pressure had leveled off at 40 mm Hg. Considerable attention was directed toward eliminating leaks in the vacuum system and lines. No significant reduction in the vacuum pressure was achieved. Periodic phone conversations with other VSC-24 users were made to discuss the situation. A determination was made by the management team after approximately 40 hours to stop the vacuum process and purge the cask with helium to further dry the basket of suspected water. During the purge, approximately 1/2 liter of additional water was removed.

The vacuum system was reconnected and progress was made in continuing to dry the basket. However, another vacuum pressure plateau at 4 mm Hg occurred. Again, considerable attention was directed toward ensuring the leak tightness of the system. The vacuum line between the cask and the vacuum pump was checked by the licensee and found to have a low spot where approximately 1/2 cup of water had accumulated. The water was removed and the pressure level soon dropped, eventually reaching a level below 2.5 mm Hg. The Technical Specifications required a vacuum pressure of ≤ 3 mm Hg to be maintained for 30 minutes. Arkansas Nuclear One had determined that 2.5 mm Hg was the correct test point based on considerations for accuracy of the test equipment. A successful vacuum pressure test was conducted on December 14, 1996, at 1 a.m at which time the pressure remained constant during the test at 2.5 mm Hg for the 30 minutes. Total elapsed time for the initial vacuum drying had been 86 hours and was considerably longer than originally expected. This long time interval of vacuum drying was due to not purging the basket with helium for a longer period of time to ensure removal of all moisture during the initial draining and the resultant low heat load of the fuel elements due to their age.

Upon completion of the initial vacuum drying, the basket was filled with helium to 22.1 psia. A helium leak test was successfully completed of the structural lid welds. The basket was vacuum dried again to approximately 2.5 mm Hg and held at that level for 30 minutes. Upon completion of the vacuum drying, the basket was backfilled with helium. The valve cover plate was welded onto the structural lid and a final helium leak and dye penetrant test was performed on the valve cover plate to verify the integrity of the weld. Both the helium test and the dye penetrant test verified the weld integrity.

h. MSB (basket) Helium Backfill Pressure

Technical Specification 1.2.8 established the required helium pressure for the final filling of the basket. A helium pressure level of 14.5 psia ± 0.5 psia, stable for 30 minutes, was required. ANO completed the final filling of the basket with helium on December 16, 1996. The pressure was verified for 30 minutes at 15 psia.

i. MSB Dye Penetrant Test of Shield and Structural Lid Seal Welds

Technical Specification 1.2.9 required the dye penetrant test of both the root pass and final pass of the inner shield lid welds and the structural lid welds be conducted. The licensee performed the dye penetrant tests using Procedure 1415.002, "Liquid Penetrant Examination," Revision 9. The procedure provided a detailed step by step process for conducting the dye penetrant test. Attachment 2 of Procedure 1415.002, "Liquid Penetrant Examination Report," was completed for each of the required tests. The dye penetrant tests of the root pass and the final pass of the shield lid were completed on December 8, 1996. The dye penetrant tests of the root pass and the final pass of the structural lid were completed on December 9 and 10, 1996, respectively.

j. Time Limit for Draining the MSB (basket)

Technical Specification 1.2.10 established a time limit for the length of time water can remain in the cask. A 47-hour limit was established by the Certificate of Compliance for a cask with 24 kW of heat. For casks with less than 24 kW, a formula was provided for determining the time limit.

Arkansas Nuclear One Procedure 1302.28, "Fuel Selection Criteria for Dry Cask Storage," Revision 1, provided an equation for determining the maximum time limit between the installation of the shield lid and the removal of water from the basket. The ANO equation was conservative compared to the equation provided in the Technical Specifications. The ANO equation included the current temperature of the spent fuel pool water. Based on the equation in the Technical Specification, ANO was allowed 217 hours to drain the basket. The equation used by ANO from their procedure established a limit of 146 hours. The shield lid was placed over the cask on December 7, 1996, at 9:10 p.m. The water was removed from the basket on December 10, 1996, at 7:40 a.m., a total of 58 hours and 30 minutes elapse time, well within the time limit established by both the Technical Specifications and ANO.

k. Minimum Temperature for Moving the MSB (basket)

Technical Specification 1.2.13 established the minimum temperature allowed for movement of the basket while inside the storage cask. This minimum temperature authorized is specified as 0 degrees Fahrenheit.

The storage cask, while loaded with the basket, was moved on a rail car from the train bay to the ISFSI pad. This was a distance of approximately 200 yards. The basket was loaded into the storage cask on December 17, 1996, at 3 a.m. Together the basket and storage cask were moved the following day to the ISFSI and placed in the assigned location on the ISFSI pad on December 18, 1996, at 5 p.m. Ambient temperature during the movement of the cask was approximately 32 degrees Fahrenheit. The ambient temperature during movement remained well above the minimum requirement of the Certificate of Compliance throughout the operation.

l. Minimum Temperature for Lifting the MTC (transfer cask)

Technical Specification 1.2.14 established a minimum temperature in which the basket and transfer cask can be lifted and moved. This minimum temperature was 40 degrees Fahrenheit.

The basket and transfer cask were partially emersed in the spent fuel pool water during the welding and vacuum drying. The temperature of the basket remained at slightly over 100 degrees Fahrenheit. During the lifting of the basket and transfer cask from the cask loading pit and onto the refueling floor, the ambient temperature was approximately 65 degrees Fahrenheit. This temperature remained relatively constant throughout the time the basket and transfer cask were on the refueling floor. When the basket and transfer cask were moved to the train bay and the basket was lowered into the storage cask, the minimum temperature in the train bay was 79 degrees Fahrenheit, well above the 40 degrees Fahrenheit limit.

m. Handling Height

Technical Specification 1.2.15 required that the storage cask not be handled at a height greater than 80 inches. The storage cask was moved between the ISFSI pad and the train bay on a specially built rail car. While on the rail car, the storage cask was 17 inches above ground. Once the storage cask was placed on the ISFSI pad, the maximum distance above grade was approximately 32 inches. At no time throughout the movement and storage of the cask was the storage cask placed in a configuration that approached the 80- inch limit.

1.3 Conclusion

The Technical Specifications established in the VSC-24 Certificate of Compliance had been incorporated into the ANO procedures and were complied with during the process of loading, moving, and storage of the first VSC-24 cask at ANO.

2 Radiological Controls (60855)

2.1 Inspection Scope

The acts of loading and moving the spent fuel to the ISFSI provided the potential for both contamination and radiological exposures of personnel. A major part of this inspection focused on the radiological controls established by the licensee for the dry cask storage program and the efforts implemented toward minimizing exposures. Considerable inspector time was directed toward observation of radiation protection personnel performing health physics duties and the review of radiological records generated during the work efforts.

2.2 Observations and Findings

The licensee provided thermoluminescent dosimetry (TLD) sensitive to both gamma and neutron radiation to all personnel involved with the dry cask storage program. The dosimetry was the same type used for normal operating plant radiation exposure monitoring.

When an individual was required to enter a "neutron dosimetry" area, extra controls were placed on the use of the dosimetry. A special TLD that had been "pre-exposed" to a known neutron fluence was used. This was required to "overcome" the deadband associated with neutron monitoring response of the TLDs. Workers were cautioned against performing any work in areas, other than in neutron dosimetry areas, where exposure to gamma radiation was likely. This precaution was due to the dosimeters not being able to discriminate between gamma radiation from external sources and the neutron-gamma reaction within the TLD chips.

Exposure rates throughout the loading and moving of the spent fuel were considerably less than estimated by the Safety Analysis Report. These lower than expected doses were primarily due to the age of the fuel. Almost no neutron exposures were encountered during the dry cask storage project, and the general beta-gamma exposure rates around the cask loading pit and spent fuel pool during the loading of the basket were below 1 mR/hr. The total manrem dose to load the 24 fuel assemblies was 16 mrem. During the welding of the shield lid and the structural lid, the transfer cask and basket were suspended over the cask loading pit with approximately half of the transfer cask and basket extending down into the water in the cask loading pit. Water was also left inside the basket to provide shielding. Exposure rates on the top of the transfer cask were 6 mR/hr on contact. General radiation levels in the area were less than 1 mR/hr. The total dose received during the welding process by the welding personnel and support personnel was 48 mrem.

Water samples taken from the basket that were used to determine the boron levels were also counted for radioisotope content. The primary isotopes detected and their approximate concentrations were:

- Co-58 5.7×10^{-4} µCi/ml
- Co-60 4.2×10^{-5} µCi/ml
- Cs-134 8.8×10^{-5} µCi/ml
- Cs-137 3.9×10^{-4} µCi/ml

Surveys of the work areas consisted of area radiation surveys as well as contamination surveys. Area radiation surveys consisted of both beta-gamma monitoring using an ion chamber type detector and neutron monitoring using Bonner spheres. The Bonner spheres were used to determine the neutron fluency/energy level to determine the appropriate neutron dose rates.

Contamination controls were strictly enforced by the health physics personnel in the area around the cask loading pit. Good contamination control practices by the licensee's staff were observed. When the basket and transfer cask were first removed from the cask loading pit, the basket and transfer cask were contaminated due to contact with the spent fuel pool water and the deposits from the spent fuel assemblies that had become dislodged during insertion of the fuel into the basket sleeves. Large area cleaning of the transfer cask with rags resulted in the rags reading up to 100 mR/hr contact with minor contamination events early in the process. However, the licensee applied adequate controls to minimize the potential for reoccurrence. After the initial minor contamination events, no personnel, area, or airborne contamination problems were encountered.

2.3 Conclusion

The ANO health physics organization implemented a strong health physics program to control radiation exposures and contamination during all cask loading and moving operations. All ANO personnel involved with the dry cask storage program were observed by the inspectors to fully cooperate with the radiological controls and demonstrated individual commitment to complying with the implemented radiation work permit requirements for the project.

3 Fuel Loading Difficulties (60855)

3.1 Inspection Scope

A critical activity for the dry cask storage program was the safe handling and moving of the spent fuel from the spent fuel pool racks to the basket located in the cask loading pit. Twenty-four fuel assemblies had to be moved and safely inserted into the sleeves in the basket without damaging the fuel assemblies. The ANO Unit 1 fuel assemblies, being slightly larger than the Unit 2 fuel assemblies, fit tightly into the basket sleeves. This inspection included direct observation of fuel movement activities and numerous interviews with personnel involved with the selection and movement of the fuel assemblies.

3.2 Observations and Findings

The first fuel assembly was placed into the basket on December 4, 1996, at 9:30 p.m. Loading of all 24 fuel assemblies was completed December 6, 1996, at 5:43 a.m. Loading of the first basket took approximately 32 hours.

Preselected spent fuel assemblies had been moved to a special area in the Unit 1 spent fuel pool. The basket and transfer cask were placed in the cask loading pit, adjacent to the spent fuel pool. The cask loading pit was separated from the spent fuel pool by a removable gate, filled with borated water, and the gate removed to allow transfer of the spent fuel assemblies from the spent fuel pool to the basket.

During the loading process, difficulty was encountered with inserting the fuel assemblies into the sleeves in the basket. This problem occurred with approximately half of the fuel assemblies. When the basket was originally placed in the cask loading pit, the orientation of the basket sleeves had not been aligned precisely with the racks in the spent fuel pool, the alignment being off by approximately 8 degrees. When a fuel assembly was transported from the fuel rack to the basket, it had to be rotated slightly to align with the sleeves in the basket. This alignment was being directed by an individual on the refueling floor looking through 30 feet of water in the cask loading pit while providing directions to the refueling machine operator. This was more of a challenge than anticipated and resulted in a significant amount of time involved to orient the fuel assemblies. The spent fuel racks had a tolerance of approximately 1/8 inch on each side. The sleeves in the basket had a tolerance of half that amount, or 1/16 inch on each side. This reduced tolerance made insertion of the fuel assemblies into the sleeves a precise maneuver. As a result of the difficulty in completing this task, ANO purchased a submersible camera system for use in future fuel movements.

Once the fuel assembly was aligned with the sleeve, the initial insertion still presented a problem on some of the fuel assemblies. The first part of the fuel assembly that was inserted into the basket sleeve was the lower end fitting. The lower end fitting was a sturdy component which supports the ends of the fuel rods.

The lower end fitting and the upper end fitting were the widest parts of the fuel assembly. The force in which a fuel assembly can be lowered into the sleeve is limited by the potential for damage to the fuel rod spacer grids. The spacer grids are located at various locations along the length of the assembly and hold the fuel rods in place. To prevent damage to the spacer grids during insertion, an underload limit is placed on the crane during loading. A fuel assembly is lowered in place by a cable. When the process starts, the cable is holding the weight of the fuel assembly plus the weight of the grappling section of the fuel loading machine. This combined weight was 2050 lbs. An indicator on the refueling machine provided a readout of the weight. As the fuel assembly was lowered into the sleeve, the weight would reduce if the fuel assembly started to bind or become stuck in the sleeve. The underload setting would stop the insertion process to prevent the entire weight of the fuel assembly from forcing the fuel assembly into the sleeve and damaging the spacer grids. The underload setting was determined to be 1810 pounds, which was 240 pounds under the total weight of the assembly and grappler.

The underload setting stopped the loading process on several fuel assemblies during the first few inches of the insertion when the lower end fitting would start to bind. The assembly would then be lifted slightly and insertion attempted again. This lifting and re-insertion process would typically be required several times. The licensee plans to reevaluate the process for inserting the fuel assemblies and determine if a lower underload setting may be more appropriate for the initial few inches of the insertion process prior to a grid strap encountering the top of the sleeve.

The problem with the Unit 1 fuel being inserted into the sleeves is not expected with the Unit 2 fuel. The Unit 2 fuel is slightly smaller than the Unit 1 fuel.

During the placement of the fuel assemblies into the basket, the water around the top of the basket unexpectedly became cloudy. To correct this problem, the licensee used an underwater filter system to improve the water clarity. Clouding of the water appeared to have been primarily the result of deposits flaking off the fuel assemblies as they were being inserted into the sleeves in the basket. The clouded water dissipated after the basket was loaded and a thermal flow was caused by the fuel elements in the basket. The clouding of the water around the basket did not appear to be a problem other than visibility while loading fuel assemblies into the basket sleeves. This problem was easily resolved with the use of the underwater filtration system.

3.3 Conclusion

Difficulties were experienced during the early phases of fuel loading when the licensee inserted the fuel assemblies into the sleeves in the basket. These difficulties, in part, resulted from not having the basket sleeves aligned with the spent fuel racks, by the use of an extremely conservative underload setting on the spent fuel crane for the first few inches of the insertion process, and by the unexpected clouding of the water around the top of the basket. The licensee identified and implemented adequate corrective actions for each of the problems.

4 Carbo Zinc Coating Interactions with Borated Water (60855)

4.1 Inspection Scope

As a result of the Point Beach incident, considerable attention had been directed toward the interaction of the carbo zinc coating with the borated water in the spent fuel pool. Two primary effects of this interaction are the generation of hydrogen gas and the formation of a zinc precipitate. During the loading of the basket with spent fuel and the welding of the shield lid, special attention was directed by the inspectors toward these two concerns.

4.2 Observations and Findings

Samples were taken of the spent fuel pool water and water in the cask loading pit on a regular basis starting on December 4, 1996, at 7:20 p.m. Fuel movement began on December 4, 1996, at 8 p.m. When the clouded water was observed during the fuel movement activities, several special samples were taken during December 5 and 6, 1996. No abnormal levels of zinc, iron, or boron were measured in the samples.

Hydrogen monitoring was conducted by ANO during the welding of the root pass weld on the shield lid. An air flow was created by the licensee under the shield lid to prevent hydrogen build-up. This air flow was monitored by a hydrogen explosives meter. This meter was set to alarm at 10 percent of the Lower Explosive Limit (LEL) for hydrogen. During the root pass welding, no hydrogen was detected. Arkansas Nuclear One continued to monitor for hydrogen on successive welds of the shield lid as a precaution and to determine if any hydrogen would be detected throughout the process. Prior to the third weld pass of the shield lid, a reading of 4 percent of the LEL was detected, which increased to 7 percent, then dropped back to zero within a couple of seconds. This was determined by the licensee to be a gas bubble of hydrogen being detected. No other indications of measurable hydrogen were noted during the welding of the shield lid.

4.3 Conclusion

Hydrogen monitoring during the welding of the shield lid was effectively implemented by ANO. On one occasion, a measurable level of hydrogen was observed for a short period of time. The maximum hydrogen level reached during welding activities was 7 percent of the LEL for a couple of seconds. The continuous flow of air under the shield lid during the welding effort was effective in preventing hydrogen build-up and potential explosion problems.

5 Safety Evaluations, Nonconformances and 10 CFR 72.48 Reviews (37001)

5.1 Inspection Scope

The licensee can only make changes from an original design for a cask or ISFSI as described in the cask's Safety Analysis Report if a safety evaluation is performed to ensure that the changes do not constitute an unreviewed safety question, a significant increase in occupational exposure, or a significant unreviewed environmental impact. If the changes involve components not described in the Safety Analysis Report, a nonconformance process can be used. This inspection included a review of selected 10 CFR 72.48 evaluations, Condition Reports, and nonconformance reports related to the VSC-24 casks used at ANO.

5.2 Observations and Findings

The following documents were reviewed during this inspection:

- Entergy Operations Incorporated ANO Procedure 1022.039, Revision 0, "Ventilated Storage Cask 10 CFR 72.48 Reviews"
- Entergy Operations Incorporated ANO Procedure 1000.104, Revision 13, "Condition Reporting and Corrective Actions"
- ANO Condition Report CR-C-96-0149, dated June 27, 1996
- ANO 10 CFR 72.48 Evaluation, "Use of Kevlar Slings in Lieu of Wire Rope for MSB Lift"
- ANO 10 CFR 72.48 Evaluation, "Address Confirmatory Action Letter"
- ANO 10 CFR 50.59 Evaluation. "Addition of Dry Fuel Storage and ANO Compliance with NUREG-0612"
- Nonconformance Report (NCR) # MTC-1-05, dated March 12, 1995
- March Metal Fab, Inc., IR #: 5059-007, dated April 25, 1995
- NCR #: MTC-1-09, dated March 27, 1995
- NCR #: MTC-1-12, dated May 12, 1995
- NCR #: YOKE-01, dated April 12, 1995

Arkansas Nuclear One Condition Report CR-C-96-0149 involved the acceptance of a nonconformance without performing a 10 CFR 72.48 safety evaluation. The nonconformance involved a swagelok well cavity opening for the MSB-03 shield lid which had been drilled to a slightly larger diameter than specified in the job order. The job order specification was based on the description in the Safety Analysis Report.

Arkansas Nuclear One had dispositioned this nonconformance under Procedure 1000.104, "Condition Reporting and Corrective Actions" instead of Procedure 1022.039, "Ventilated Storage Cask 10 CFR 72.48 Reviews." The licensee believed that the use of Procedure 1000.104 was appropriate because this was a one time acceptance of the nonconformance and was not a permanent change to the design of the cask. Only permanent changes to the VSC-24 system were reviewed under the criteria of 10 CFR 72.48, and Procedure 1022.039.

Procedure 1000.104 does not require the same type of evaluation as specified in 10 CFR 72.48. In all, there were 12 nonconformances which individually resulted in one-time changes to the design criteria in the Safety Analysis Report and were dispositioned using Procedure 1000.014 instead of performing a 10 CFR 72.48 review required under Procedure 1022.039.

Five of these nonconformances involved the transfer cask, which was used during the loading of the first cask at ANO. These were:

- <u>NCR # MTC-1-05, dated March 12, 1995</u>

  Drawing dimensions and tolerances were not met for the door on the bottom of the MTC.

- <u>March Metal Fab, Inc IR #: 5059-007, dated April 2 , 1995</u>

  The 5/8-inch weld between the bottom of the cask and the rail assembly of the doors did not meet the code required profile.

- <u>NCR #: MTC-1-09, dated April 27, 1995</u>

  The weld between the bottom plate of the MTC and the MTC vessel was aligned out of tolerance.

- <u>NCR #: MTC-1-12, dated May 12, 1995</u>

  The door at the bottom of the cask stuck during initial testing. A few thousands of an inch was milled off the door where the sticking occurred.

- <u>NCR #: YOKE-01, dated April 12, 1995</u>

  The inside diameter of the cask lifting yoke is specified as 0.5 inch greater than the outside diameter of the transfer cask. The outside diameter of the transfer cask is 82.5 inch. However, the inside diameter of the lifting yoke was only 82.31 instead of the required 83 inch. The size of the yoke was increased by reducing the thickness of the shim material.

For each of these examples, the licensee performed engineering safety evaluations that determined that the components could be used as-is. However, the nonconformances were not evaluated in accordance with Procedure 1022.039, and no documentation was completed to verify that an unreviewed safety question, as defined in 10 CFR 72.48, did not exist.

The NRC's position concerning the requirements in 10 CFR 72.48(a) & (b) is that any change to the VSC-24 system, whether it is a one time change to one cask or a design change to several casks, requires a 10 CFR 72.48 evaluation if the change affects the design as described in the Safety Analysis Report. The Safety Analysis Report for the VSC-24 includes component drawings, specifications, and tolerances for the basket, transfer cask, and storage cask. Use of any components that deviates from these design specifications requires a 10 CFR 72.48 review. Use of the transfer cask during the loading of the first basket at ANO, without completing the required safety evaluations for the nonconformances on the transfer cask is a violation of 10 CFR 72.48 and will be tracked as 50-313/9625-01.

5.3 Conclusions

The Safety Analysis Report for the VSC-24 cask included design specifications and detailed drawings. Any deviation from these design specifications, whether the deviation was a one-time nonconformance that affected an individual cask or a permanent design change that affected several casks, required an evaluation as specified in 10 CFR 72.48. Several nonconformances were accepted for the ANO transfer cask without completing a 10 CFR 72.48 safety evaluation. This was identified as a Severity Level IV violation.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

R. Bement, RP/Chemistry Manager
E. Coolen, Quality control
J. Dosa, Licensing Engineer
N. Finney, Nondestructive Examination Specialist
G. Hettel, Unit 2 Senior Reactor Operator
R. Kellar, HLW Project Manager
C. Larrison, Unit 1 Mechanical Maintenance Supervisor
J. McWilliams, Modification Manager
T. Nickels, RP Specialist
J. Priore, Reactor Engineer
R. Rego, Unit 1 Operator
J. Smith, Radiation Protection Operations Supervisor
D. Williams, Engineer
P. Williams, Nuclear Safety Analysis

INSPECTION PROCEDURES USED

60855	Operations of an ISFSI
37001	10 CFR 50.59 Safety Evaluation Reviews

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-313/9625-01	VIO	Failure to complete 72.48 safety evaluations

LIST OF ACRONYMS USED

ANO	Arkansas Nuclear One, Units 1 & 2
ANSI	American National Standard Institute
CAL	Confirmatory Action Letter
dpm	disintegrations per minute
ISFSI	Independent Spent Fuel Storage Installation
LEL	Lower Explosive Limit
mR/hr	milliRoentgen/hour
mrem/hr	millirem/hour
MSB	Multi-assembly Sealed Basket
MTC	Multi-assembly Transfer Cask
MWd	Megawatt Day
MTU	Metric Tons Uranium
NRC	Nuclear Regulatory Commission
ppm	parts per million
psia	pounds per square Inch (absolute)
scc/sec	standard cubic centimeters/second
VCC	Ventilated Concrete Storage Cask
VSC	Ventilated Storage Cask (System)