UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

July 10, 2012

Brian J. O'Grady, Vice President
Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION – NRC INSPECTION OF THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION - INSPECTION REPORT 05000298/2010009 AND 07200066/2010001

Dear Mr. O'Grady:

This inspection report covers the inspection of your Independent Spent Fuel Storage Installation (ISFSI) conducted between September 13, 2010, and February 10, 2011. This included the team inspections between September 13, 2010, and October 21, 2010, to observe your dry cask storage program preoperational demonstrations and the loading of the first canister. The team inspections consisted of three inspection trips and involved a total of nine NRC inspectors. An exit was conducted on October 21, 2010, to review the overall results of the team inspections. On February 9 - 10, 2011, a reactive inspection was performed in response to the partial draindown of the neutron shield of the transfer cask loaded with Canister No. 2. An exit was conducted of the findings for that inspection on February 10, 2011. Subsequent to these inspections, the NRC inspection team performed an extensive in-office review of licensing documents and various dry cask storage program documents to verify that all requirements and licensing conditions had been incorporated into your procedures and programs consistent with the Transnuclear NUHOMS Certificate of Compliance No. 1004, Technical Specifications, and the NUHOMS Updated Final Safety Analysis Report.

The dry fuel storage program implemented at the Cooper Nuclear Station was found to be comprehensive and fully developed. The first loading of a dry fuel storage cask was safely controlled and successfully performed. The NRC inspection team reviewed a broad range of topical areas related to programs required to successfully move spent fuel from your spent fuel pool to dry cask storage at your ISFSI storage pad. The inspections consisted of an examination of selected procedures, observations of dry-run training activities, interviews with personnel, and observations of the first cask loading. The inspections examined activities conducted under your license as they relate to public health and safety to confirm compliance with the Commission's rules and regulations, orders, and with the conditions of your license.

Based on the results of these inspections, the NRC has determined that one Severity Level IV violation occurred. The Severity Level IV violation related to the inadvertent draining of water from the transfer cask's neutron shield tank. The reduction in shielding resulted in an increase in the dose rates in the local work area. The NRC is treating this violation as a noncited

violation, consistent with Section 2.3.2 of the NRC Enforcement Policy because the issue was entered into your corrective action program, you took effective and immediate corrective actions, and the event was not repetitive or willful. This issue is discussed in the attached inspector notes under the Category: Operations and the Topic: Unintentional Draindown of Transfer Cask.

If you contest the violation or the significance of the violation, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 1600 East Lamar Blvd, Arlington, TX 76011-4511. In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal, privacy or proprietary information so that it can be made available to the public without redaction.

Should you have any questions concerning this inspection, please contact the undersigned at (817) 200-1191 or Mr. Vincent Everett at (817) 200-1198.

Sincerely,

/RA/

D. Blair Spitzberg, Ph.D., Chief
Fuels Safety and Decommissioning Branch

Dockets: 50-298, 72-66
Licenses: DPR-46

Enclosure:
Inspection Report Nos.:
05000298/2010009,
07200066/2010001

Attachments:
1. Supplemental Inspection Information
2. Loaded Casks at the Cooper ISFSI
3. Cooper ISFSI Inspection 72-66/10-01 Inspector Notes

cc w/Enclosure and Attachments: Electronic Distribution

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298, 72-66
Licenses:	DRP-46
Report Nos.:	05000298/2010009 and 07200066/2010001
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station Independent Spent Fuel Storage Installation (ISFSI)
Location:	Brownville, NE 68321
Dates:	September 13 - 17, 2010, Program Review Inspection September 27 - October 2, 2010, Dry Run Demonstrations Inspection October 11 - 21, 2010, First Cask Loading February 9 - 10, 2011, Reactive Inspection
Team Leader:	Vincent Everett, Senior Inspector, RIV Fuels Safety and Decommissioning Branch
Inspectors:	Lee Brookhart, Health Physicist, RIV Gerald Schlapper, Health Physicist, RIV Abin Fairbanks, Reactor Inspector, RIV Leonard Willoughby, Senior Project Engineer, RIV Jim Pearson, Senior Trans & Storage Safety Inspector, NMSS Jack Parrott, Safety Inspection Engineer, NMSS John Vera, Structural Engineer, NMSS Pamela Longmire, Project Manager, NMSS
Approved By:	D. Blair Spitzberg, Ph.D., Branch Chief Fuels Safety and Decommissioning Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Cooper Nuclear Station
NRC Inspection Report 05000298/2010009 and 07200066/2010001

The NRC conducted an extensive review and evaluation of the Cooper Nuclear Station's program for the safe handling and storage of spent nuclear fuel at their Independent Spent Fuel Storage Installation (ISFSI). This included observation of the preoperational training demonstrations, loading of the first cask, and response to an incident involving the unintentional partial draining of the shield water in the transfer cask. The Cooper Nuclear Station had selected the Transnuclear Standardized NUHOMS Horizontal Modular Storage System, approved under Certificate of Compliance No. 1004, as their ISFSI design. The version of the NUHOMS systems used at the Cooper Nuclear Station included the 61BT dry shielded canister (DSC), the HSM-202 horizontal storage module (HSM), and the OS197H transfer cask. Cooper had constructed an ISFSI pad to hold fifty-two horizontal storage modules (HSMs), each containing one canister loaded with sixty-one spent fuel elements. The ISFSI was licensed by the NRC under the general license provisions of 10 CFR Part 72. The licensee planned to load eight canisters for placement on the ISFSI pad during their first loading campaign in 2010/2011. The first canister loading was observed by the NRC in October 2010.

The inspections conducted by the NRC of Cooper's dry cask storage project included a comprehensive evaluation of the licensee's compliance with the requirements in the Transnuclear NUHOMS Certificate of Compliance No. 72-1004 and Technical Specifications, Amendment 9; the Updated Final Safety Analysis Report (UFSAR), Revision 10; the NRC's Safety Evaluation Report, Amendment 9; and 10 CFR Part 72. A program review was conducted the week of September 13, 2010, by a team of NRC inspectors who performed an in-depth review of the required ISFSI programs. Cooper developed a preoperational test plan which consisted of two demonstrations performed during the weeks of February 23, 2009, and September 27, 2010, that were observed by the NRC. Certificate of Compliance No. 1004, Technical Specification 1.1.6, listed eight specific demonstrations that were required of the licensee. Technical Specification 1.1.6, Part 6 (DSC sealing, vacuum drying, and cover gas backfilling operations) and Part 7 (opening a DSC) were demonstrated on February 23, 2009, and were documented in Inspection Report 72-66/09-001, dated August 28, 2009, (NRC ADAMS Accession No. ML092430509). The remaining demonstrations, Parts 1 through 6 and Part 8, were demonstrated at the Cooper nuclear facility during the week of September 27, 2010. Twenty-one technical areas were reviewed during the inspections including such topical areas as crane design, crane inspection, crane operations, drying/helium backfill, fuel verification, radiological programs, quality assurance, heavy loads, training, welding, and others. Subsequent to the site visits, an extensive in-office review was performed of documents provided by the Cooper staff. This effort involved the review of several thousand pages of reports, procedures, calculations, training documents, test results, personnel qualification records, safety evaluations and condition reports to support the conclusion that the licensee had developed and implemented a comprehensive program to support ISFSI activities.

During the inspections, the licensee successfully demonstrated the operation of equipment and the implementation of procedures required by the license to safely load a canister and place it at the ISFSI. The NRC review of numerous documents provided by the licensee concluded that the licensing requirements related to dry cask storage had been adequately incorporated into Cooper's programs and procedures. During the various preoperational demonstrations and first loading, the Cooper workers demonstrated a comprehensive knowledge of the technical

requirements related to the loading and operations of an ISFSI. Cooper's first cask was placed on the ISFSI pad on October 21, 2010.

Details related to the technical areas reviewed during this inspection are provided as Attachment 3 "Cooper ISFSI Inspection 72-66/10-01 Inspector Notes" to this inspection report. The following provides a summary of the various "categories" listed in Attachment 3.

Crane Design

- The drum safety devices and hoist holding brake design of the crane met the requirements of NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants." The drum retaining devices were designed such that a failure would not cause the main hoist drum to disengage. The holding brakes were designed for 125% capacity. A single failure would leave two holding brakes operable for stopping and controlling drum rotation.

- The licensee had addressed the Part 21 notifications from Whiting Corporation, the manufacturer of the crane, concerning the support bolts in the hoist unit gear case.

- The reactor building walls that supported the reactor building crane were verified to be capable of holding the 108 ton rated load of the crane under normal operating conditions and seismic events. The maximum critical load plus operational and seismically induced pendulum and swinging load effects on the crane were taken into consideration.

- The licensee had evaluated their current 108 ton crane against the criteria in NUREG 0554 and found the crane to meet the criteria for a single failure proof crane. This included safety systems such as overload protection and two-block protection.

- The licensee had the ability to manually lower the load and manually move the bridge and trolley if an emergency occurred causing a loss of power to the crane. These provisions were described in the licensee's procedures.

- The 108 ton crane used two separate ropes, one right hand lay rope and one left hand lay rope. Calculations showed that with the maximum load (including static and inertia forces) on the system, the ropes did not exceed the 10% breaking strength limit specified in NUREG 0554, Section 4.1 or the wire rope breaking strength criteria from NRC Branch Technical Position APCSB 9-1.

Crane Inspection

- The 108 ton crane was inspected annually as required by ASME B30.2. The inspection included checking for deformed, cracked, or corroded members; cracked or worn sheaves and drums; worn, cracked or distorted pins, bearings, shafts, gears, rollers, etc.

- Prior to using the reactor building crane, a crane inspection was performed by the licensee at the beginning of each shift. The inspection used the guidance in ASME B30.10 for the hook inspection and ASME B30.2 for the crane/support structure and wire rope inspection.

- A crane performance test was completed at the Cooper site after the crane modifications were completed to increase the capacity of the crane from 100 tons to 108 tons. The tests included hoist raising/lowering at all speeds, trolley travel in both directions at all speeds, bridge travel in both directions at all speeds, and testing of all safety devices.

Crane Licensing Basis

- The NRC inspectors reviewed the licensee's basis for removing the reactor building crane 70 ton weight restriction from the Technical Requirements Manual. The crane had originally been rated at 100 tons. Due to several nonconformances with NRC Branch Technical Position APCSB 9-1, a 70 ton limit was placed on the crane in License Amendment No. 35. The licensee made numerous crane modifications and completed a new analysis that demonstrated that the 70 ton restriction could be removed. The licensee completed a 50.59 screening to remove the 70 ton limit.

- NRC inspectors reviewed the analyses performed by Burns and Roe and Stevenson Associates which demonstrated that no additional modifications to the reactor building structure were necessary to support the new 108 ton rated load of the crane.

- The licensee had performed upgrades and modifications to load bearing components on the crane to up-rate the crane from 100 to 108 tons. The crane modifications included replacement of the variable frequency drive for the main hoist motor, replacement of two lower load cell connection pins, increasing the size of two welds on the equalizer bar support plate, and enlarging the end holds for the rope anchors in the two vertical end plates of the equalizer assembly.

Crane Load Testing

- Cooper's newly modified 108 ton crane completed a 100% dynamic load test prior to fuel loading activities. A load of 109.1 tons was used during the test, which included raising and lowering at all speeds and movement of the bridge/trolley at various speeds in all directions. All safety devices and limit switches were tested with no load on the hook.

- Cooper's newly modified 108 ton crane was statically loaded to approximately 125% of the rated load. A load of 133 tons (123.1% of the rated load) was used during the test which included raising and lowering at all speeds and movement of the bridge/trolley in all directions over the longest distance possible while varying the speeds. The licensee was restricted by ASME B30.2 (Revision 1976) to not load test the crane greater than 125% the rated load.

- The maximum weight lifted by the 108 ton crane occurred when the transfer cask was lifted from the spent fuel pool, loaded with fuel, and filled with water. This maximum weight was calculated to be 106.2 tons, which was within the crane's capacity.

Crane Operation

- The licensee's procedures required brake checks prior to lifting a loaded cask and specified a minimum travel height when moving the cask.

- The Whiting crane manufacturer's recommended preventative maintenance program was incorporated into the licensee's maintenance program.

- Qualification requirements for the Cooper Nuclear Station crane operators were consistent with the requirements listed in ASME B30.2.

Drying/Helium Backfill

- Vacuum drying requirements related to canister dryness levels were incorporated into the licensee's procedures consistent with the requirements in Technical Specification 1.2.2.

- The licensee had established provisions to ensure canisters with heat loads greater than 17.6 kW included vacuum drying time limits consistent with Technical Specification 1.2.17.

- The 61BT canisters were required by the licensee's procedures to be backfilled with helium to a pressure of 1.5 psig to 3.5 psig. This was consistent with Technical Specification 1.2.3.a. The first canister was backfilled to a pressure of approximately 2.5 psig.

Emergency Planning

- The new ISFSI was located within the protected area of the operating reactor and was incorporated into the Part 50 reactor emergency plan. The emergency plan included emergency action levels for classifying an emergency at the ISFSI consistent with the emergency classification scheme used at the reactor. Offsite support for an emergency at the ISFSI was provided under the same agreements established for the Part 50 reactor emergency plan. On October 8, 2010, a drill was conducted that incorporated an emergency event during the simulated movement of a loaded canister to the ISFSI.

Fire Protection

- A detailed fire and explosion hazards evaluation was performed by the licensee to evaluate nearby hazards to the haul path and ISFSI pad. Over twenty-seven potential fire and explosion hazards were identified and evaluated. Limits were calculated for the maximum quantity of explosive or flammable material allowed for varying distances from the haul path or ISFSI. The licensee stated that as more horizontal storage modules are loaded, a fire barrier may be required between the ISFSI and the craft change building.

Fuel Selection/Verification

- Only intact fuel was being selected for loading during the first loading campaign.

- The licensee required independent verification that the correct fuel assembly was selected prior to placement into the canister. The process used an underwater camera to determine if the correct assembly was selected for transfer. The independent verification process for the first canister loaded was observed by the NRC inspectors.

- Fuel selected for storage in the first canister was compared to Technical Specification 1.2.1 and associated tables and found to meet the requirements related to maximum enrichment, burnup, decay heat, and cooling time for storage in the NUHOMS casks.

- Material balance and inventory records were generated and maintained in accordance with 10 CFR 72.72. The ISFSI and the canister were added to the procedures as item control areas. Records were generated showing the location in the canister of each spent fuel assembly by serial number.

General License Requirements

- Changes to the site related to the construction and operation of the ISFSI were evaluated in accordance with 10 CFR 72.48 and 10 CFR 50.59 requirements.

- The licensee performed an analysis that demonstrated that no real individual member of the public beyond the owner controlled area would receive a dose in excess of the limits in 10 CFR 72.104 from a fully loaded ISFSI at Cooper. The ISFSI was located within the plant's owner controlled area such that the nearest real individual member of the public would be at least 800 meters away. Dose calculations from the fully loaded ISFSI at 800 meters away projected 0.07 mrem/yr. When added to the projected dose from reactor plant operations, the total dose was calculated to be 1.13 mrem/yr, which was below the 10 CFR 72.104 limit of 25 mrem/yr.

- The Transnuclear Certificate of Compliance and Updated Final Safety Analysis Report (UFSAR) had been reviewed by the licensee to verify that the design basis for the Transnuclear cask system and the conditions and requirements in the Certificate of Compliance and UFSAR were met.

- The licensee evaluated the bounding environmental conditions specified in the UFSAR and technical specifications against the conditions at the site. This included: floods, seismic events, lightning, snow, normal and abnormal temperatures, and tornados/high winds. The flooding conditions for the ISFSI at Cooper were bounded by the 15 feet/second flood velocity and 50 foot high flood limitations specified in Technical Specification 1.1.1.4. The elevation of the ISFSI base mat lies above the elevation of the probable maximum flood for the site.

- The licensee performed an evaluation of the Part 50 reactor programs that could be impacted by the addition of an ISFSI. The evaluation included the radiation protection program, emergency planning program, quality assurance program, training program, reactor technical specifications, and the Part 50 license. Revisions to the programs to incorporate the ISFSI were identified and implemented. None of the changes required an amendment to the plant's Part 50 operating license or technical specifications.

- Cooper had developed specific ISFSI procedures for controlling all work associated with cask handling, loading, movement, surveillance, maintenance, and testing. In addition, procedures developed for the Part 50 reactor programs were being adequately applied to the ISFSI program, where applicable.

Heavy Loads

- The licensee's heavy loads procedural requirements related to "prior-to-use" inspection of the transfer cask trunnions, lift yokes, and transfer cask interior/exterior surfaces were consistent with UFSAR Section 4.5.1.

- Safe load paths for the heavy lifts within the licensee's reactor building were identified and incorporated into the licensee's procedures. Temperature and height restrictions for loading, transporting, and unloading operations were incorporated as limitations in the applicable procedures.

- The transfer cask lifting trunnions were load tested to 300% of the maximum load prior to use of the transfer cask.

Nondestructive Examination

- The visual and liquid penetrant examination procedures implemented all the applicable requirements from ASME Section III, Section V, and the Certificate of Compliance in regards to nondestructive examination of welds.

- Helium leak rate tests were performed on the inner top cover seal weld consistent with the acceptance standards specified in the Certificate of Compliance and ANSI N14.5. The helium leak testing equipment used during the first loading was verified to meet the minimum sensitivity level specified in ANSI N14.5.

Operations

- Requirements related to preoperational inspections and maintenance of equipment were incorporated into the licensee's procedures and were being implemented in accordance with the frequencies specified in the UFSAR.

- During the loading of the first canister beginning October 13, 2010, the NRC provided 24-hour coverage of the loading operations for all the critical tasks. This included fuel movement, heavy lifts, radiation surveys, welding of the lid, vacuum drying, helium backfill, transportation of the canister to the ISFSI, and insertion of the canister into the horizontal storage module. The first canister was placed on the ISFSI pad October 21, 2010.

- The thermal performance of the first cask placed in service was assessed and a letter submitted to the NRC dated November 15, 2010, in compliance with Technical Specification 1.1.7.

- The licensee implemented daily temperature reading of the in service horizontal storage modules using thermocouples. The licensee's procedures adequately incorporated Technical Specifications 1.3.2 and 1.2.8 requirements to ensure the thermal conditions would not exceed concrete and fuel clad temperature criteria.

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- Requirements for hydrogen monitoring during welding of the cask lid had been incorporated into the procedures. Procedures required welding to be stopped if hydrogen levels reached 60 percent of the lower explosive limit.

- On November 3, 2010, the licensee made a required 24-hour report to the NRC concerning loading operations for the second cask in which an unintentional partial draining of the transfer cask's neutron shield occurred while loaded with a canister of spent fuel. The event occurred in the reactor building railroad airlock area and resulted in an increase in dose rates in the area. The failure to follow procedures had resulted in the opening of the drain lines for the neutron shield, allowing a partial draindown of the water in the shield. As a result of the incident, the licensee identified a violation of Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI" and opened several non-compliance reports to correct the violation and prevent recurrence. Because the violation was a Severity Level IV violation, was self-identified and put into the licensee's corrective action program, the issue is being treated by the NRC as a non-cited violation (NCV). [A detailed event description can be found in the attached inspector notes under the Category: Operations and the Topic: Unintentional Draindown of Transfer Cask.]

Preoperational Test

- The licensee successfully completed all the required dry run demonstrations specified by Technical Specification 1.1.6. This included loading a mock fuel assembly into a canister; welding, drying, and backfilling the canister; and transporting the canister between the reactor building and the ISFSI pad. A weighted canister was used to demonstrate heavy load activities inside the reactor building, transport between the reactor building and the ISFSI, insertion of the canister into a horizontal storage module, and movement back into the reactor building for unloading purposes.

Quality Assurance

- The licensee's quality assurance program previously approved by the NRC for use under the Part 50 reactor license was being used for the Part 72 ISFSI license.

- All instruments used for the first cask loading that required calibration were within their calibration dates.

- The corrective action program established measures to ensure conditions adverse to quality were promptly identified and corrected. Condition reports associated with the ISFSI activities and the reactor building crane were selected for review. The issues identified in the condition reports had been adequately resolved.

- The UFSAR identified structures, systems, and components that were important to safety and categorized each item into one of three levels (A, B, or C) based on safety significance. The licensee incorporated Transnuclear's safety designations into their classification procedure used to determine the level of quality control to place on the items.

- The licensee had incorporated the Part 72 activities into their quality assurance program. Audits and surveillances of ISFSI activities had been performed. Issues were placed in the corrective action system for resolution.

Radiation Protection

- The station ALARA program was applied to the dry cask storage loading operations. ALARA controls were implemented throughout the loading campaign to reduce unnecessary exposures and keep personnel exposures low.

- Radiation controls and contamination controls were included in the licensee's procedures for the various ISFSI activities. This included contamination surveys of the transfer cask, canister lid, transfer cask annulus area, and radiation surveys of the horizontal storage modules.

- Surveys of the first loaded horizontal storage module confirmed compliance with Technical Specification 1.2.7. All exposure rates were less than 1 mrem/hr at three feet from the surface of the horizontal storage module. The exposure rates at the vents at the bottom were 25 mrem/hr on contact and 10 mrem/hr at 30 cm.

- Calculations had been performed by the licensee to demonstrate compliance with the public exposure levels established in 10 CFR 72.104 and 72.106. This included doses due to direct radiation from the ISFSI and doses during a postulated accident.

- Neutron monitoring was performed during cask loading operations. The licensee had accounted for the change in the neutron energy spectrum that would occur when the water was removed from the canister.

- Procedural steps and cautions had been developed for taking a sample of the air inside the canister as part of the process to remove a canister lid to unload a canister. The licensee recognized that the radiation levels from the sample could be in the R/hr range if damage to the fuel cladding had occurred.

Records

- The licensee was maintaining the ISFSI records in their quality-related records system consistent with the requirements of 10 CFR 72.212 and 10 CFR 72.234. The records were required to be maintained for the life of the ISFSI.

Slings

- The appropriate slings were used by the licensee for various lifting activities. Dual and redundant slings with a load rating twice the sum of the static and dynamic loads were required for critical lifts, meeting the criteria in NUREG 0612.
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- The licensee's sling maintenance program met the requirements of ASME B30.9 for sling inspections and removal of slings from service.

Special Lifting Devices

- The lift yoke utilized by the licensee for lifting activities met the requirements of ANSI N14.6 for initial load testing, annual maintenance, and preoperational inspections.

Training

- The licensee had established a training program for ISFSI operations. Only trained and certified personnel were allowed to operate equipment and controls that had been identified as important to safety in the UFSAR in accordance with 10 CFR 72.190.

- The training material incorporated into the licensee's program met the training requirements listed in the UFSAR for canister preparation and handling, fuel loading, transfer cask preparation and handling, and transfer trailer loading. The training program also included the requirement for generalized training on the applicable regulations, standards, and engineering related to passive cooling, radiological shielding and structural characteristics of the ISFSI.

Welding

- The licensee's procedures incorporated the requirements of ASME Section III for consumption of tack welds and required weld lengths. No unacceptable welds were identified on the first canister.

ATTACHMENT 1: SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel

S. Anderson, Projects Field Coordinator
T. Barker, Quality Assurance Manager
S. Bebb, Document Management Supervisor
S. Bray, SFT Campaign Manager
L. Covington, Fuels
M. England, ISFSI Project Manager
B. Hasselbring, Senior Reactor Operator
A. Jacobs, ISFSI Procedure Coordinator
B. Kirkpatrick, Licensing Specialist
J. Long, Senior Reactor Operator
C. Mayer, FME Monitor
D. Montgomery, Emergency Preparedness Manager
K. Mowery, Nuclear Support
S. Rezab, Radiation Protection Health Physicist
K. Schroeder, NDE Examiner
N. Shubert, ISFSI Project Controls
R. Slama, Maintenance Shop Specialist
T. Stevens, ISFSI Project Manager
C. Sunderman, Maintenance & Technical Training Superintendent
T. Tinker, ISFSI Engineering Technician
B. Victor, Licensing Engineer
D. Werner, Operations Training
D. Williams, Project Engineer
B. Wolken, Civil Engineer Supervisor

Contractors

S. Atwater, Rocky Mountain Engineering
S. Bantz, Bartlett Radiation Protection Technician
B. Devins, Areva Fuel Handler
J. Hargett, Areva Shift Manager
J. Keesee, Areva Fuel Handler
S. Lampe, Bartlett Radiation Protection Technician
K. Limoge, TriVis Level II NDE Specialist
G. Miaris, TriVis Level II NDE Specialist
R. Nurney, Bartlett Radiation Protection Technician
R. Reinstadtler, ACECO Site Representative
E. Sanders, TriVis Cask Loading Supervisor
K. Schroeder, NDE Specialist
K. Thompson, Areva Fuel Handler
K. Woods, Consultant

INSPECTION PROCEDURES USED

IP 60854.1	Preoperational Testing of ISFSIs at Operating Plants
IP 60856	Review of 10 CFR 72.212(b) Evaluations
IP 60857	Review of 10 CFR 72.48 Evaluations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NCV	72-66/1001-001	Draining of Transfer Cask

Discussed

None

Closed

NCV	72-66/1001-001	Draining of Transfer Cask

LIST OF ACRONYMS

Abs	absolute
ACECO	American Crane and Equipment Corporation
AEC	Atomic Energy Commission
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
AWS	automated welding system
BTP	Branch Technical Position
Btu/hr	British thermal unit per hour
BWR	boiling water reactor
C	Celsius
cc/sec	cubic centimeters per sec
CED	Change Evaluation Document
CFR	Code of Federal Regulations
cm/sec	centimeter per second
CMAA	Crane Manufacturers Association of America
CMTR	certified materials test report
CNS	Cooper Nuclear Station
CoC	Certificate of Compliance
CR	condition report
DOE	Department of Energy
dpm	disintegrations per minute

DSC	dry shielded canister
EAL	emergency action level
EE	Engineering Evaluation
EEIPS	extra extra improved plow steel
ENSA	Equipos Nucleares S.A.
EPD	electronic personnel dosimeter
EPIP	Emergency Plan Implementing Procedure
F	Fahrenheit
FCN	FSAR Change Notice
fpm	feet per minute
ft/sec	feet per second
g	gravity
GE	General Electric
GWD/MTU	Giga Watt Day per Metric Ton Uranium
HMSLD	helium mass spectrometer leak detector
Hz	hertz
HSM	horizontal storage module
ICA	item control area
ISG	Interim Staff Guidance
ISFSI	Independent Spent Fuel Storage Installation
ITS	important to safety
IWRC	independent wire rope core
kg	kilogram
kW	kilowatt
LBDCR	licensing basis document change request
lbs	pounds
LCO	limiting condition for operation
LRL	left regular lay
m/sec	meters per second
MCNP	Monte Carlo N-Particle
mrem	MilliRoentgen Equivalent Man
MSL	mean sea level
NEI	Nuclear Energy Institute
NEMA	Nebraska Emergency Management Agency
NCV	non-cited violation
NDE	non-destructive examination
NOUE	notice of unusual event
NPPD	Nebraska Public Power District
NRC	Nuclear Regulatory Commission
NITS	not important-to-safety
OCA	owner controlled area
OSL	optically stimulated luminescence
OSHA	Occupational Safety & Health Administration
PM	preventative maintenance
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PT	liquid penetrant exam
QA	quality assurance
QAPD	quality assurance program description
RCT	radiological control technicians

rpm	revolutions per minute
RRL	right regular lay
RWP	radiation work permit
SSC	safety systems and components
SER	Safety Evaluation Report
SNM	special nuclear material
SSE	safe shutdown earthquake
SWP	Special Work Permit
TC	transfer cask
TLCO	Technical Limiting Condition of Operation
TLD	thermo-luminescent dosimetry
TRM	Technical Requirements Manual
TS	technical specification
TSR	Technical Surveillance Requirement
U-235	Uranium 235
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
VDS	vacuum drying skid

ATTACHMENT 2: LOADED CASKS AT THE COOPER ISFSI

LOADING ORDER	DSC SERIAL No.	HSM No.	DATE ON PAD	HEAT LOAD (kW)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
1	CNS61B-007-A	HSMA-1A	10/21/10	11.3256	37,505	3.390	0.700
2	CNS61B-005-A	HSMA-2A	10/29/10*	11.3230	37,522	3.390	0.608
3	CNS61B-006-A	HSMA-3A	11/24/10	11.2859	37,513	3.390	0.760
4	CNS61B-003-A	HSMA-4A	12/03/10	11.2675	37,748	3.390	0.630
5	CNS61B-001-A	HSMA-1B	12/10/10	11.2645	37,507	3.390	0.554
6	CNS61B-008-A	HSMA-2B	12/16/10	11.2592	37,741	3.390	0.513
7	CNS61B-002-A	HSMA-3B	01/03/11	11.2417	37,738	3.390	0.566
8	CNS61B-004-A	HSMA-4B	01/13/11	11.2031	37,736	3.390	0.566

NOTES:
- Heat load (kW) is the sum of the heat load values for all spent fuel assemblies in the cask
- Burn-up is the value for the spent fuel assembly with the highest individual discharge burn-up
- Fuel enrichment is the spent fuel assembly with the highest individual “initial” enrichment per cent of U-235

SPECIAL NOTE: Canister CNS61B-005-A (second loaded) was inserted into the HSM on October 29, 2010, removed October 31, 2010 and reinstalled on November 11, 2010

COOPER ISFSI INSPECTION 72-66/10-01

INSPECTOR NOTES

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COOPER ISFSI INSPECTION 72-66/10-01
INSPECTOR NOTES

Category: Crane Design **Topic:** Drum Safety Devices

Reference: APCSB 9-1 (1975) Section B.3.k Issued 1975

Requirement: The load hoisting drum should be provided with structural and mechanical safety devices to prevent the drum from dropping, disengaging from its holding brake system, or rotating, should the drum or any portion of its shaft or bearing fail.

Observation: Report REP-20881-001, Section 4.2 "Drum Supports" provided a description of how the drum safety devices on the reactor building crane were engineered to prevent a load drop. The drum retaining devices had close fitting retainers at their hubs or supports, which ensured that a shaft or bearing failure would not allow the main hoist drum to disengage from the drum and pinion gear mesh and hence disengage from the hoist braking system.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) Letter from Dennis Ziemann, NRC to J. M. Pilant, Nebraska Public Power District entitled "Issuance of Amendment 35 to Cooper Nuclear Station Facility Operating License No. DPR-46," dated February 28, 1977, (c) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007

Category: Crane Design **Topic:** Hoist Control Brake Operation

Reference: APCSB 9-1 (1975) Section B.3.m Issued 1975

Requirement: The minimum hoist braking system should include one power control brake (not mechanical or drag brake type) and two mechanical holding brakes. The holding breaks should be activated when power is off and should be automatically tripped by mechanical means on overspeed to the full holding position if a malfunction occurs in the electrical brake controls. Each holding brake should be designed to 125% - 150% of maximum developed torque at the point of application.

Observation: Report REP-20881-001, Sections 4.9 "Hoist Braking System" and Section 5.1 "Braking Capacity" provided an explanation of the hoist braking system to demonstrate compliance with the Branch Technical Position requirements. Section 4.9 stated that the main hoist control system was provided with dynamic braking through the flux vector drive and two mechanical shoe-type holding brakes. The main hoist's two shoe-type holding brakes were on the high speed shafting to hold the load during normal operation. A single failure, such as a dynamic brake failure, would leave the two holding brakes operable for stopping and controlling drum rotation. The holding brakes in the main hoisting system were applied when power was off or when a drum overspeed occurred. Section 5.1 stated that the holding brakes located on each motor were automatically applied when power was off. Each holding brake of the main hoist was designed with a minimum capacity of 125% of the torque developed during the hoisting operation at the point of brake application.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Crane & Equipment Corporation

Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007

Category: Crane Design **Topic:** Hoist Holding Brake Operation

Reference: APCSB 9-1 (1975) Section B.3.m Issued 1975

Requirement: The minimum design requirements for braking systems that will be operable for emergency lowering after a single brake failure should be two holding brakes for stopping and controlling drum rotation. Provisions should be made for manual operation of the holding brakes. Emergency brakes or holding brakes which are to be used for manual lowering should be capable of operation with full load and at full travel and provide adequate heat dissipation. Design for manual brake operation during emergency lowering should include features to limit the lowering speed to less than 3.5 fpm.

Observation: The main hoist of the licensee's reactor building crane used two shoe-type holding brakes on the high speed shafting to hold the load during normal operation. A description of the braking system was provided in Report REP-20881-001, Section 4.9 "Hoist Braking System." The main hoist control system was provided with dynamic braking through the flux vector drive. The brake wheels were mounted on an extension of each motor pinion input shaft. A single failure, such as power loss, would leave two holding brakes operable for emergency lowering. The shoe brakes on the main hoist can be manually operated to lower a load in the event of hoisting equipment failure. Each holding brake was provided with adequate capacity to stop and hold the full load, but not excessive to cause damage to hoisting machinery. The main hoist drum overspeed system was provided with a speed indicator, mounted on the trolley deck, that can be utilized to display the lowering speed during emergency operations. The bridge and trolley brakes included a manual release lever to permit manual emergency operation. Attachment points on the trolley and bridge allowed for manual manipulation to move the load to a safe area to set down the load or repair the crane.

Procedure 7.6.1, Section 9.1 "Crane Loss of Power Recovery" established provisions for manual operation of the holding brakes in the event of an emergency. Procedure 7.6.1, Step 9.2.6.2 stated: "Since each brake is capable of stopping and holding a full load, one brake can be held open while the other brake is manipulated. Do not allow any part of the brake to exceed 250 degrees F. Alternate the north and south brakes in controlling the load to keep brakes from overheating." Step 9.2.6.3 stated: "Control the descent by allowing the brakes to close if downward motion becomes too rapid. Judge speed of the load by the readout of the tachometer. Do not allow the hoist to exceed the normal hoist creep speed rate of 0.72 feet/minute, which corresponds to the motor shaft rotation speed of 60 revolutions/minute (rpm)."

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24, (c) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007

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Category: Crane Design **Topic:** NRC Information Notice 2003-20

Reference: NRC Information Notice 2003-20 October 22, 2003

Requirement: On January 29, 2003, Whiting Corporation submitted a 10 CFR 21 report (Event Notification No. 39545 and Part 21 Notification No. 2003-002-00) to the NRC. The notification alerted the NRC to a problem with the Whiting #25 Hoist Unit (Gear Case). When lifting a load near its nominal rating, the stresses in one or possibly two internal support bolts in this assembly may be significantly over the design allowable stress. These bolts connect the gear case housing to an open frame that supports bearings and other components in the gear train. If a bolt failed, the open frame might deform, affecting gear alignment. Whiting Corporation stated that a 50% reduction in the rated hoist capacity would allow continued use of the crane without compromising design safety factors. This issue applied to Cooper's 100 ton reactor building crane.

On February 12, 2003, Whiting Corporation issued a follow up notice (Letter #301-816-5151) to the NRC. The notification reported a similar issue with the bolts associated with the Whiting #10 Hoist Unit (Gear Case). Whiting Corporation stated that a 20% reduction in the rated hoist capacity would allow continued use of the crane without compromising design safety factors. This issue applied to the 5 ton auxiliary hoist on the Cooper reactor building crane.

Observation: Cooper Nuclear Station addressed the Part 21 notification issued by Whiting Crane concerning the bolts by replacing the suspect bolts on both the 100 ton hoist and 5 ton auxiliary hoist. Parts Evaluation No. 4292709 was issued by Cooper concerning their evaluation. For the 100 ton hoist, the original bolts were equivalent to an ASTM A307, Grade B bolts. The replacement bolts were procured with a certificate of compliance to SAE Grade 8 for the bolt material. The replacement bolt material had a minimum tensile strength of 150,000 psi. The original bolt material had a minimum tensile strength of 60,000 psi. For the 5 ton auxiliary hoist, the original bolt was equivalent to an ASTM A307, Grade B bolt. The replacement bolt material exceeded SAE Grade 8 specifications. The replacement bolt material had a minimum ultimate strength of 170,000 psi. The original bolt had a minimum ultimate strength of 68,000 psi.

Documents Reviewed: (a) NRC Information Notice 2003-20 "Derating Whiting Cranes Purchased Before 1980," dated October 22, 2003 [NRC Adams Accession No. ML032960205], (b) Letter to the NRC from Whiting Corporation (Letter #301-816-5151) entitled "10 CFR 21 Notification Dated 1/29/2003," dated February 12, 2003 [NRC Adams Accession No. ML030520271], (c) Cooper Parts Evaluation Number 4292709 "Replacement of #25 Hoist Unit Housing Bolts and One Housing Bolt on #10 Housing Hoist Unit," Revision 1, (d) 10 CFR Part 21 Notification 2003-002-00 "Whiting Corporation #25 Hoist Unit (Gear Case) Support Bolt Overstress," dated January 29, 2003, (e) Event Notification 39545 "Whiting Crane Part 21 Notification Concerning Hoist Unit Bolts," dated January 29, 2003

Category: Crane Design **Topic:** Overload Protection

Reference: NUREG 0554, Section 4.5 Published May 1979

Requirement: The complete hoisting system should have the required strength to resist failure if the hosting system should "two block" or if load "hang-up" should occur. As an alternative,

the protective control system to prevent the hoisting system from two-blocking should include, as a minimum, two independent travel-limit devices of different design and activated by a separate mechanical means. These devices should de-energize the hoist drive motor and the main power supply.

Observation: Cooper's reactor building crane's main hoist employed the alternate method of redundant travel limit switches to prevent a two-blocking event. Report REP-20881-001, Section 4.5 "Design Against Two Blocking" discussed the protection system on the reactor building crane. A hang-up event was prevented by overweight switches and a load cell. The main hoist contained a primary rotary travel limit switch and a secondary lever-operated power limit switch. The travel limit switch on the drum shaft sensed both the upper and lower positions of the load block travel and stopped the motion by de-energizing the hoist controls when tripped. The secondary power limit switch was tripped by the lower block and directly disconnected power to the hoist motor when tripped. The travel limit switch activated first to prevent continued raising of the load. The power limit switch was set above the travel limit switch in the event the travel limit switch malfunctioned, in which case the power limit switch would trip and turn off the power to the crane, preventing any further hoisting motion. Both the travel limit switch and power limit switch were tested per Procedure REP-20881-014, Section C during the load testing when the crane was re-rated from 100 tons to 108 tons. Additionally, the travel limit switch was tested daily per the daily crane checkout Procedure 7.6.1, Attachment 3 "Crane Operator Daily Inspection Checklist," Step 1.2.3.

Documents Reviewed: (a) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979, (b) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007, (c) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24, (d) American Crane & Equipment Corporation Procedure REP-20881-014 "Site Functional and Load Test Procedure for Reactor Building Crane Controls Upgrade," Revision 0

Category: Crane Design **Topic:** Provisions For Manual Operation

Reference: NUREG 0554, Sections 3.4; 4.9 Published May 1979

Requirement: A crane that has been immobilized because of failure of controls or components while holding a critical load should be able to hold the load or set the load down while repairs or adjustments are made. This can be accomplished by inclusion of features that will permit manual operation of the hoisting system and the bridge and trolley transfer mechanisms by means of appropriate emergency devices.

Observation: Procedure 7.6.1 included provisions to manually operate the crane. The main hoist of the reactor building crane used two shoe-type holding brakes on the high speed shafting to hold the load during normal operation. During a power loss, the holding brakes automatically closed, but could be manually operated for emergency lowering. The bridge and trolley brakes also automatically closed on a loss of power. These brakes included a manual release lever to permit manual emergency operation to reposition the bridge and trolley. Procedure 7.6.1, Section 9.1 "Crane Loss of Power Recovery" provided instructions for operating the crane during a loss of power. Section 9.2 "Main Hoist Recovery" provided instructions for lowering the load. Step 9.2.6.2 stated: "Since each brake is capable of stopping and holding a full load, one brake can be held open

while the other brake is manipulated. Do not allow any part of the brake to exceed 250 degrees F. Alternate the north and south brakes in controlling the load to keep brakes from overheating." An infrared thermometer was listed as one of the required tools in Section 9.2.1.1. Step 9.2.6.3 stated: "Control the descent by allowing the brakes to close if downward motion becomes too rapid. Judge speed of the load by the readout of the tachometer. Do not allow the hoist to exceed the normal hoist creep speed rate of 0.72 feet/minute, which corresponds to the motor shaft rotation speed of 60 revolutions per minute (rpm)." Section 9.3 "Bridge Recovery" provided instructions for manually releasing the bridge brakes to allow movement of the bridge using a ratcheting come-along. Section 9.4 "Trolley Recovery" provided instructions for releasing the trolley brakes and moving the trolley using a ratcheting come-along. If power was lost, the tornado latches would lock onto the rail due to loss of hydraulic pressure and would have to be manually raised and secured. Step 9.3.3 "Tornado Latches Disengagement (Bridge)" and Step 9.4.2.3 "Tornado Latches Disengagement (Trolley)" discussed the need to open the latches to move the trolley and bridge. The licensee stated that during maintenance activities, there had been situations where they had to manually raise and secure the tornado latches. This process was relatively simple.

Documents Reviewed: (a) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979, (b) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24, (c) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007

Category: Crane Design **Topic:** Seismic Events During Cask Movement

Reference: APCSB 9-1 (1975) Section B.1.c Issued 1975

Requirement: The crane should be classified as seismic Category I and should be capable of retaining the maximum design load during a safe shutdown earthquake, although the crane may not be operable after the seismic event. The bridge and trolley should be provided with means for preventing them from leaving their runways with or without the design load during operation or under seismic loadings.

Observation: The reactor building crane was capable of stopping and holding the load during a seismic event. The reactor building was a seismic class 1 structure listed in the Cooper Updated Safety Analysis Report (USAR), Section XII-2.1.2 "Class I Structures and Equipment." The crane was identified in the USAR, Section XII-2.3.5.1.8 "Cranes" as being designed in accordance with the criteria for Class I earthquake loading. NUREG 0554, Section 2.5 "Seismic Design" stated "The crane should be designed and constructed in accordance with Regulatory Position 2 of Regulatory Guide 1.29. Regulatory Guide 1.29 defined the criteria for Seismic Category I structures, systems, and components. Regulatory Position 2 required that the crane be designed and constructed such that it's failure would not reduce the functioning of any plant feature designated as seismic Category I that was required to withstand a safe shutdown earthquake and remain functional. Items meeting Regulatory Position 2, such as the crane, did not have to remain functional after the seismic event. Report REP-20881-001, Section 2.5 "Seismic Design" reviewed the features for the reactor building crane designed to withstand a safe shutdown earthquake and determined that the crane met the criteria of Position 2 of Regulatory Guide 1.29.

Calculation NEDC 07-077 and DP Engineering Calculation CNS-07-01-CALC-01 evaluated the reactor building steel superstructure to determine if it was adequate to hold the 108 ton load suspended on the crane during a seismic event. The calculations also evaluated the reactor building for the 125% load test which would include a static load of 141 tons. The evaluation considered the safe shutdown earthquake and the effects from pendulum action. The calculations evaluated the reactor building crane in accordance with NUREG-0554 and Regulatory Guide 1.29 and determined that the existing reactor building steel superstructure was structurally adequate. The licensing basis for the reactor building crane limited the girders to a stress of 0.9 Sy (See Amendment 33, Section 1.1 "Reactor Building Crane-Description of Modifications"). The licensee's Design Calculation NEDC 09-023 demonstrated that the licensing basis was met and Change Evaluation Document (CED) 6028740, Tab 6, Section 2.1 was used by the licensee to demonstrate that the re-rated crane (carrying the 108 ton rated load) will accommodate the loadings of an operating basis earthquake without exceeding AISC stress limits for the girders, and a safe shutdown earthquake without exceeding 0.9 Sy stress limits for the girders. CED 6028740 also stated that the original Burns & Roe calculation for crane live loads on the reactor building superstructure remained bounding.

The USAR, Section XII-2.3.5.1.8 "Cranes" stated that the reactor building crane was equipped with hold-down lugs to maintain stability and prevent release from the rails in the event of an earthquake or tornado. The reactor building crane and supporting steel were designed in accordance with the criteria for Class 1 earthquake loading. The crane was analyzed with maximum operating live loads. USAR Section XII 2.3.3.2.4 "Tornado Loads - Additional Considerations" stated that the reactor building crane and the supporting steel superstructure were designed to withstand tornado loading. Both the crane and columns were designed for tornado loadings with a rack and locking device such that the crane would be locked to the supporting structure which would prevent the wind loads from pushing the crane off the crane runways. Bridge and trolley wheels were double flanged and equipped with electrically activated sparing set brakes. In the event of loss of power or when the crane was not under operator control, the design provided for spring activated brakes which would lock the wheels firmly in place. Positive wheel stops and bumpers were provided in order to prevent the trolley and bridge from leaving the rails in the unlikely event of brake failure.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) Engineering Design Calculation (NEDC) 07-077 "Seismic Evaluation of Reactor Building with Loaded Reactor Building Crane," Revision 0, Status 2, dated July 13, 2010 including DP Calculation CNS-07-01-CALC-01, Revision 4, (c) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24, (d) Engineering Design Calculation (NEDC) 09-023 "Whiting Corporation-Crane Re-Rate Design Report," Revision 0, (e) Change Evaluation Document (CED) 6028740 "Cooper Nuclear Station Reactor Building Crane Re-Rate," dated June 25, 2010, (f) Condition Report CR-CNS-2008-07968 "Some Structures of the Reactor Building Crane May Not Be Adequately Designed to Mitigate Damage During a Design Basis Event," initiated October 29, 2008, (g) Condition Report CR-CNS-2009-02495 "Reactor Building Crane Seismic Analysis," initiated March 26, 2009, (h) Letter from J. M. Pilant, Nebraska Public Power District to L. M. Muntzing, NRC entitled

“Amendment No. 33 to Operating License DPR-46 for Cooper Nuclear Station AEC Docket No. 50-298,” dated May 3, 1974 [NRC ADAMS Accession No. ML12144A086], (i) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979, (j) US NRC Regulatory Guide 1.29 "Seismic Design Classification," Revision 3, (k) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007

Category: Crane Design **Topic:** Seismically Induced Load Swing

Reference: APCSB 9-1 (1975) Section B.1.c Issued 1975

Requirement: The design rated load plus operational and seismically-induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge.

Observation: The maximum critical load plus operational and seismically induced pendulum and swinging load effects on the crane were taken into consideration. Report REP-20881-001, Section 2.5 "Seismic Design" provided an evaluation of the Cooper crane against the criteria of NUREG 0554. The report concluded that the pendulum effect due to horizontal seismic input and swinging load effects were deemed insignificant based on similar analyses performed previously. This conclusion was consistent with the American Society of Mechanical Engineers (ASME) NOG-1, Table 4153.7-1 "Crane Load Conditions for Seismic Analysis, Static, and Dynamic Load Cases" which stated in Footnote 2 that increases in horizontal load due to pendulum effect need not be considered due to the relatively small displacement of the load. As such, the rated load was only applied in the vertical direction in the seismic analysis of the crane.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007, (c) American Society of Mechanical Engineers (ASME) NOG-1 “Rules for Construction of Overhead and Gantry Cranes,” Revision 2004, (d) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979

Category: Crane Design **Topic:** Single Failure Proof

Reference: NUREG 0554, Section 1.0 Published May 1979

Requirement: When reliance for the safe handling of critical loads is placed on the crane system itself, the system should be designed so that a single failure will not result in the loss of the capability of the system to safely retain the load.

Observation: The Cooper Nuclear Station reactor building crane met the requirements in NUREG 0554 and NUREG 0612 to be considered a single failure proof crane. In September 2007, Cooper received two independent analyses from different companies that verified that the crane conformed to the NUREG 0554 and NUREG 0612 requirements as a single failure proof crane for 100 tons. The two reports were Engineering Ltd. Evaluation DP 07-002 and American Crane and Equipment Corp. Report REP-20881-001. To upgrade from 100 tons to 108 tons, Whiting Corporation, which was the manufacturer of the crane, performed an evaluation to determine if the crane would maintain the required margins to perform as a single failure proof crane at 108 tons. The

Whiting evaluation, issued as Design Calculation (NEDC) 09-023, identified all under-design margins as a result of lifting a 108 ton load and recommended various structural changes. These design changes were forwarded to Cooper for review and implementation. All recommended changes from Whiting Corporation were implemented through Change Evaluation Document (CED) 6028740. This change evaluation document was used to perform the structural upgrades considered necessary to maintain the crane at the design structural performance margins with the higher hook load. CED 6028740 replaced the variable frequency drive for the main hoist motor, replaced two lower load cell connection pins with higher strength material pins, increased the size of two six inch welds on the equalizer bar support plate from 1/2 inch to 5/8 inch, installed girder stiffening bars, and replaced the main hoist cable with larger diameter wire ropes. The Whiting evaluation demonstrated that with the recommended modifications, the crane's structural members would maintain a substantial margin to yield strength when loaded to 108 tons. The results of the safety factor calculation showed that all structural members remained within the Crane Manufacturers Association of America (CMAA) Guide #70 allowable stresses. Safety factors for non-structural single-failure proof components of the load path were 5:1 or greater and non-structural non single-failure proof components of the load path were 10:1 or greater. Additionally, CED 6028740 included an evaluation demonstrating that the associated modifications met the requirements of NUREG 0612, Appendix C "Modifications to Existing Cranes."

Documents Reviewed: (a) Engineering Ltd Evaluation DP 07-002 "Cooper Nuclear Station Reactor Building 100-Ton Crane Evaluation of Single Failure Proof Crane," dated September 18, 2007, (b) Engineering Design Calculation (NEDC) 09-023 "Whiting Corporation-Crane Re-Rate Design Report," Revision 0, (c) Change Evaluation Document (CED) 6028740 "Cooper Nuclear Station Reactor Building Crane Re-Rate," dated June 25, 2010, (d) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007, (e) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979, (f) NUREG 0612 "Control of Heavy Loads at Nuclear Power Plants," issued July 1980, (g) Crane Manufacturers Association of America (CMAA) Guide #70 "Top Running and Gantry Type Multiple Girder Electric Overhead Traveling Cranes," released 1971,

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Category: Crane Design **Topic:** Two-Block Protection

Reference: APCSB 9-1 (1975) Section B.3.J Issued 1975

Requirement: The mechanical and structural components of the hoisting system should have the required strength to resist failure should "two-blocking" or "load hang-up" occur during hoisting. The location and type of mechanical brakes and controls should provide positive and reliable means to stop and hold the hoisting drums for these occurrences. The hoisting system should be able to withstand the maximum torque of the driving motor, if a malfunction occurs and power to the driving motor cannot be shut off at the time of load hang-up or two-blocking.

Observation: The crane was equipped with two upper limit switches that protected the crane from hoisting the load block into itself. The limit switches were discussed in Report REP-20881-001, Section 4.5 "Design Against Two-Blocking." NUREG 0554, Section 4.5 "Design Against Two-Blocking" allowed for dual limit switches as an alternative to the

mechanical and structural components of the hoisting system having the required strength to resist failure during a two-blocking incident. Two-blocking protection and load hang up protection are discussed in these Inspector Notes under the Category: Crane Design and the Topic: Overload Protection.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Crane & Equipment Corporation Report REP-20881-001 "NUREG 0554/0612 Compliance/Safety Analysis Report," dated September 27, 2007

Category: Crane Design **Topic:** Wire Rope Breaking Strength

Reference: APSCB 9-1 (1975), Section B.3.e Issued 1975

Requirement: The stress in the lead line to the drum hoisting at the maximum design speed with the design rated load should not exceed 20% of the manufacturers rated strength of the rope. The static stress in rope (load is stationary) should not exceed 12.5% of the manufacturer's rated strength.

Observation: The current wire rope on the reactor building crane met the breaking strength requirements of Branch Technical Position APSCB 9-1. The two ropes currently on the crane, replaced in 2009, were 1-1/4 inch, Python Power 9V, 9 x 25, Extra Extra Improved Plow Steel (EEIPS) wire ropes as described in Whiting's Crane Re-Rate Design Report for the Cooper Nuclear Station, page iv of xi. One rope was a right regular lay (RRL) and one was a left regular lay (LRL). Several documents, including the purchase order for the ropes described the ropes as 9 x 19. Python provided information to the licensee that the rope purchased in 2009 was a 9 x 19 classification of rope. The 9 x 19 classification includes both 9 x 19 ropes and the 9 x 25 ropes. The smaller ropes supplied by Python were 9 x 19. For sizes above 7/8 inch, the ropes were 9 x 25. The 1-1/4 inch rope used at Cooper was a 9 x 25. There were also documents in the licensee's files that referenced the current rope as 9S versus 9V. Python informed the licensee that the "V" designator was used in the U.S. to designate the swaged construction and in Europe, the "S" is used. The Cooper rope was 9V. Breaking strength test certificates provided with the two ropes by Unirope listed the test results for the break strength as 217,600 lbs (108.8 tons) for the RRL rope and 216,100 (108.05 tons) for the LRL. The break test certificates provided by Unirope did not indicate whether the tests pulled the ropes to failure or pulled the ropes to the values listed in the break test certificates and then stopped the test. As such, the value of 108 tons will be used as the break strength for the calculations below. There were 6 reeves of rope on the crane resulting in 12 parts of rope between the crane and the hook. For the static load, each part of rope will be carrying the weight of 1/12 the load. The load consisted of both the rated load of 108 tons, the weight of the block at 6 tons, and the weight of the rope at 2.85 tons for a total of 116.85 tons. The weight of the block was from the Whiting Re-Rate Design Report, Section 1 "Trolley Design Evaluation." The weight of the rope was based on the Python web page which listed the rope weight as 3.33 pounds/foot. The rope weight would be 855 feet x 2 ropes x 3.33 pounds/foot = 5,694 pounds = 2.85 tons.

Branch Technical Position APCSB 9-1, Section B.3.e had two criteria that must be met concerning the stresses on the rope: one criterion related to the static stress and one related to the lead line stress. The Branch Technical Position stated that the static stress

in the rope (load is stationary) should not exceed 12.5% of the manufacturers rated strength. The 12.5% equated to a minimum safety factor of 8 (100/12.5 = 8). License Amendment No. 33 for the Cooper Nuclear Station provided the original calculations in Section 4.2 “Lead Line Safety Factors.” The static factor of safety was calculated using the rope breaking strength times 12 ropes divided by the weight of the load. For the new rope, this equals to (108 tons x 12 ropes)/116.85 tons = 11.09. Amendment 33 also provided a calculation that included the seismic component by using the same equation above, but adding a 1.07g factor to the load. So the new load when considering seismic would be 116.85 tons x 1.07 = 125 tons and the safety factor would be (108 tons x 12 ropes)/125 tons = 10.37. Both cases exceed the required Branch Technical Position limit of 8 for the safety factor. The requirement in NUREG 0554, Section 4.1 “Reeving System,” for rope safety factors was also compared to the current rope. Section 4.1 states “The maximum load (including static and inertia forces) on each individual wire rope of a dual reeving system with the maximum critical load attached should not exceed 10% of the manufacturer’s published breaking strength.” The 10% would be a minimum safety factor of 10 (100/10). A factor of 5% is reasonable for the inertial forces. The calculation to determine the safety factor would use a weight factor for the load of 116.85 tons x 1.05 = 122.7 tons. The safety factor then becomes (108 tons x 12 ropes)/122.7 tons = 10.56, and as such, meets the NUREG 0554 wire rope break strength criteria.

The second Branch Technical Position APCSB 9-1 requirement was that the stress on the rope at the maximum design speed with the rated load should not exceed 20% of the manufacturer’s rated strength of the rope. The 20% would equate to a minimum safety factor of 5 (100/20). For the stresses on the lead line, Amendment No. 33 to the Cooper license provided the equations for the lead line safety factor in Section 4.2. A factor of 0.099 as the lead line factor for 12 parts of rope was used. An explanation of the lead line factor was found in the Whiting Crane handbook on page 135 which stated “The actual maximum load in the various parts of the rope occurs in the two lead lines from the drum during hoisting and in the two lines from the equalizer sheaves when lowering. The actual load in one of these lines may be found by using the lead line factor, a function of the reeving efficiency, taken from Table 11 “Efficiency of Load Block (Double Reeved)” and multiplied by the sum of the rated load and the weight of the block.” The value in Table 11 for 12 parts of rope was 0.102. This value was consistent with the 0.099 value used in Amendment 33. The calculations in Amendment 33 to determine the lead line stress factor was the rope breaking strength divided by the weight of the load times the lead line factor. For the new rope, this equates to 108 tons/(116.85 tons x 0.099) = 9.34. This safety factor exceeded the required safety factor of 5 in Branch Technical Position APCSB 9-1. NUREG 0554 did not have an equivalent requirement for lead line safety factor.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) Whiting Corp. Customer Order No. 4200001242, Work Order No. 137091, Project No. CO9976.41 “Cooper Nuclear Station Crane Re-Rate Design Report 100/5 Ton Reactor Building Crane,” (which contained numerous sections with different dates), dated July 16, 2008, February 27, 2009, April 15, 2009 (as Revision 1), and May 22, 2009 (as Revision 1), (c) Unirope Test Certificate for Item 115859, Item #1, Reel No. 7887, 1-1/4” 855 ft Python Power 9V EEIPS, Right

Regular Lay (RRL), Bright 9-Strand High Strength Wire Rope, EEIPS, dated January 14, 2009, (d) Unirope Test Certificate for Item 115859, Item #2, Reel No. 7887, 1-1/4” 855 ft Python Power 9V EEIPS, Left Regular Lay (LRL), Bright 9-Strand High Strength Wire Rope, EEIPS, dated January 14, 2009, (e) Unirope Breaking Strength Test Certificate for Item 115859, Item #1, Reel No. 7887, 1-1/4” 15 ft Python Power 9V EEIPS, Right Regular Lay (RRL), Bright Wire Rope, EEIPS, dated January 14, 2009, (f) Unirope Breaking Strength Test Certificate for Item 115859, Item #2, Reel No. 7887, 1-1/4” 15 ft Python Power 9V EEIPS, Left Regular Lay (LRL), Bright Wire Rope, EEIPS, dated January 14, 2009, (g) Letter from J. M. Pilant, Nebraska Public Power District to L. M. Muntzing, NRC entitled “Amendment No. 33 to Operating License DPR-46 for Cooper Nuclear Station AEC Docket No. 50-298,” dated May 3, 1974 [NRC ADAMS Accession No. ML12144A086]

Category: Crane Design **Topic:** Wire Rope Specifications

Reference: APSCB 9-1 (1975), Section B.3.e Issued 1975

Requirement: The design of the rope reeving system should be dual. The wire rope should be 6 x 37 Iron Wire Rope Core (IWRC) or comparable classification. Line speed during hoisting (raising or lowering) should not exceed 50 feet/minute.

Observation: The wire rope currently on the reactor building crane was comparable to the classification described in the Branch Technical Position APSCB 9-1. Several ropes had been installed on the crane's trolley since the original installation in 1970. Prior to installation of each rope, an analysis was completed to verify that the new rope was equivalent or superior to the previous rope. The wire rope on the original trolley installed in 1970 was a 1-1/8 inch 6 x 37 Extra Flexible Improved Plow Steel (IWRC) rope as listed in the original crane bid request in Contract No. E-68-36, Section 7.3 “Hoisting Ropes” and a certification letter from Universal Wire Products, Inc. In the 1974/1975 period, a new trolley containing a new rope was purchased and installed. Documentation provided by U. S. Steel listed the rope for the new trolley as a 1-1/4 inch 6 x 37 classification monitor AAA X-lay IWRC. The right regular lay rope break tests were listed as 192,000 pounds (test 1) and 193,000 pounds (test 2). In 1984, the rope was replaced due to damage during an attempt to level the block. The 50.59 reportability analysis stated that the replacement rope was identical to the old rope. In 2002, the rope was replaced again. The need for replacement was documented in Notification 10163120 which stated that during an inspection of the cable per Work Order 4230209, rust and pitting was found on the cable. A vendor was brought in for a more thorough investigation and recommended that the cable be replaced. Procedure 3.4.1, Attachment 2 “Parts Evaluation Document Summary,” for parts Evaluation No. 10215951 provided information on the 2002 rope. The problem statement in Attachment 2 stated “The replacement cable installed on the reactor building overhead crane was not a like-for-like replacement. The description of the purchase did not match the material that was installed on the crane in 1984. The vendor manual noted that the material installed during the 1984 cable replacement was 1-1/4 inch Special Flexible Improved Plow Steel Wire Rope Cable, 6 strands, 37 wires, wire core. The actual number of wires was verified as 41. The 2002 replacement cable description on the purchase order was 1-1/4 inch Extra Extra Improved Plow Steel, 6 strand, 36 wires. The analysis of the differences between the 1984 wire rope and the 2002 wire rope discussed the variance in

the number of wires for a 6 x 37 rope which could vary from 29 to 46 actual wires. The number of wires in the 1984 rope was 41. The new 2002 rope had 36 wires. The evaluation stated that typically fatigue resistance was lowered and the rope was less flexible when the number of wires is decreased. The loss of flexibility in the case of the 2002 rope was determined to not be a problem. The slight decrease in fatigue resistance was not a concern either since the rope was regularly tested. The 2002 rope would be more resistant to abrasion. The tensile strength of the 2002 rope exceeded that of the original rope. The symbols of EEIP vs XXIP were different symbols for identifying the extra extra improved plow steel. The new rope had been listed as XXIP. The material of construction for both the old rope and the new rope from the certificates for the cables was reviewed and found to be identical materials. Both ropes were the same diameter of 1-1/4 inch. The evaluation concluded that both ropes were equivalent for use on the reactor building crane. Procedure 3.4.1, Attachment 3 "Part Evaluation Design Requirements Comparison" provided a side-by-side comparison of the two ropes to show they were equivalent. Attachment 3 showed the 1984 rope as having a tensile strength certification value of 180,600 pounds for the first test and 182,100 pounds for the second test. These values were listed on the Universal Wire Rope certificate of test as the value at which the rope broke. The certificate of conformance from Wire Rope Corp. for the new 2002 rope listed the breaking point for the rope as 183,400 pounds. A 50.59 screening was performed on the difference between the wire strands for the 1984 rope (41 wires) and the 2002 rope (36 wires) and determined that the new rope was acceptable.

In 2009 as part of the re-rating of the crane from 100 tons to 108 tons, a new rope was installed. The purchase order for the 2009 rope listed two ropes, 855 feet each, 1-1/4 inch diameter, Class 9 x 19 Python HS9V or Power 9V, EEIPS wire rope, one right regular lay and one left regular lay. The rope was more narrowly identified in the Whiting Crane Re-Rating Design Report as a Python Power 9V, Class 9 x 25 Swage Compaction EEIPS wire rope. Breaking strength test certificates provided with the two ropes by Unirope listed the test results for the break strength as 217,600 lbs (108.8 tons) for the right regular lay rope and 216,100 (108.05 tons) for the left regular lay, which exceeded the 2002 rope break values. A 100% and 125% load test was performed on the crane after all modifications were completed for the 108 ton re-rate, including installation of the new rope. The 50.59 screening of the modifications to the crane for the 108 ton re-rate, completed in accordance with Change Evaluation Document CED 6028740, found that all changes, including the replacement rope, did not adversely affect the crane design functions or reduce the safety factors associated with the crane. In conclusion, the requirement of Branch Technical Position APCSB 9-1 for the rope to be a 6 x 37 IWRC rope has been met with the new 2009 Class 9 x 25 EEIP wire rope through a series of evaluations since the installation of the new trolley and rope in 1975 to demonstrate that the new rope meets or exceeds the original requirements.

For hoisting the load, the crane's maximum rope velocity was 34.2 feet/minute with a full load. This was calculated in the Whiting Nuclear Design Survey for Purchase Order E68-36, Amendment #13, Section 3.e for the new trolley installed in 1975. This value was less than the 50 feet/minute limit in Branch Technical Position APCSB 9-1.

Documents Reviewed: (a) Contract No. E-68-36 Bid Specification "Overhead Traveling Cranes and Accessories," dated 1968, (b) Letter from Universal Wire Products to Whiting Corp. providing certification of the wire rope, dated October 22, 1970, (c) Document from U. S. Steel providing break test results for wire rope 1-1/4 inch 6 x 37 Classification Monitor AAA X-lay right regular lay IWRC rope from Reel 70017, dated June 3, 1974, (d) Nebraska Public Power District 10CFR50.59 Reportability Analysis, dated April 3, 1984 included in document package Minor Design Change Package MDC 84-044, dated April 3, 1984 (e) NPPD Notification 10163120 "Building HST-H20 Main Hoist Cable is Bad," dated May 14, 2002, (f) Procedure 3.4.1, Attachment 1 "Parts Evaluation Document Cover Sheet," Attachment 2 "Parts Evaluation Document Summary," and Attachment 3 "Part Evaluation Design Requirements Comparison," for Parts Evaluation Number 10215951, dated January 8, 2003, (g) Universal Wire Rope Products Certificate of Test for 1-1/4 inch 6 x 37 IWRC EEIP Rope Certificate of Test, dated January 12, 1984, (h) Wire Rope Corp. Certificate of Compliance for the 1-1/4 inch 6 x 36 WS RR XXIP NUC Rope, dated November 11, 2002, (i) Administrative Procedure 0.8 "10CFR50.59 and 72.48 Reviews," Attachment 3 "50.59 Screen Form" for Parts Evaluation 10215951 for the New (2002) Wire Rope," dated January 8, 2003, (j) Purchase Order 012832, Attachment 1 for American Crane and Equipment Corp., dated December 18, 2008, (k) Whiting Corp. Customer Order No. 4200001242, Work Order No. 137091, Project No. CO9976.41 "Cooper Nuclear Station Crane Re-Rate Design Report 100/5 Ton Reactor Building Crane," dated February 27, 2009, (l) Procedure 08 "10CFR50.59 and 72.48 Reviews," Attachment 3 "50.59 Screen Form," for Activity CED 6028740 "Re-Rate Reactor Building Crane from 100 Tons to 108 Tons," dated July 7, 2010, (m) Whiting Nuclear Design Survey for Purchase Order E68-36, Amendment #13, dated October 24, 1975

Category: Crane Inspection **Topic:** Crane Inspection - Annually

Reference: ASME B30.2 (1976), Section 2-2.1.3 Revision 1976

Requirement: Cranes in regular use shall be subjected to a periodic crane inspection annually during normal and heavy service and quarterly during severe service. The periodic inspection includes checking for: a) deformed, cracked or corroded members; b) loose bolts or rivets; c) cracked or worn sheaves and drums; d) worn, cracked or distorted pins, bearings, shafts, gears, rollers, locking and clamping devices; e) excessive wear on brake system parts, linings, pawls and ratchets; f) load, wind, and other indicators over their full range for any significant inaccuracies; g) gasoline, diesel, electric, or other power plants for improper performance; h) excessive drive chain sprocket wear and chain stretch, and; i) deterioration of controllers, master switches, contacts, limit switches and pushbutton stations.

Observation: The reactor building crane was inspected as required on a periodic basis of once a year. The reactor building crane was classified as a Class A, standby or infrequent service crane. Procedure 7.2.73 contained all the required inspection criteria from American Society of Mechanical Engineers (ASME) B30.2, Section 2-2.1.3 for the annual inspection. Step 4.1.4 required functionally checking the limit switches for the trolley and bridge. Step 4.1.6 required examining and testing the limit switches for the hoist. Step 4.1.7 required examining the drum. Step 4.1.8.1 required examining the bridge steel members and welds for damage, corrosion, and deformation. Step 4.1.8.2 required

checking the entire crane for loose bolts, nuts, rivets, or other fasteners. Step 4.1.10.3 required examining the sheaves for excessive wear and gouges. Step 4.1.10.4 required checking for damage in the bearing housing. Step 4.1.10.7 required checking all pins, bolts, screws and welds on the load block to ensure they were intact and not damaged or distorted. Step 4.1.11 required examining the hoist brakes for lining wear, grooved brake wheels, damage or broken shoes; examining the brake wheel rims for worn, bent or broken linkages or missing parts; and examining brake wheel surfaces for damage. Step 4.1.12 required examining brakes on the bridge and trolley for worn or bent linkage, worn linings, damaged scored, and damaged or overheated wheels. Steps 5.2 required performing electrical examinations on all control cabinets, relays, starters, loose connections or overheated wiring/terminals. These sections of the procedure were most recently completed on September 19, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Maintenance Procedure 7.2.73 "Reactor, Turbine, and Auxiliary Turbine Building Crane Examination, Maintenance, and Testing," Revision 14

Category: Crane Inspection **Topic:** Crane Inspection - Frequent

Reference: ASME B30.2 (1976), Section 2-2.1.2 Revision 1976

Requirement: Cranes in regular use shall be subjected to a visual crane inspection monthly during normal service, weekly to monthly during heavy service, and daily to weekly during severe service. The inspection points should include: (1) operating mechanisms for misadjustment interfering with proper operation - daily; (2) all limit switches should be checked, without a load on the hook, at the beginning of each work shift; (3) leakage in lines, tanks, valves, pumps, and other parts of the air or hydraulic systems - daily; (4) deformed or cracked hooks - visual inspection daily; (5) hook latches checked daily for proper operation; (6) hoist ropes including tightness of the end clamps and rope clips; (7) hoist chains, including end connections, for wear, twist, distortion links interfering with proper function, or stretch beyond manufacturer's recommendations; (8) slings, including end connections, for wear, broken wires, stretch, kinking or twisting (daily per B30.9-1971); and (9) rope reeving for noncompliance with crane manufacturer's recommendations.

Observation: The applicable sections of the American Society of Mechanical Engineers (ASME) B30.2, Section 2-2.1.2 inspection requirements were being performed on the reactor building crane at Cooper. The reactor building crane was a Class A standby or infrequent use crane. As such, the inspection requirements established for a regular use crane were not required to be implemented as often. Cooper had included most of the Section 2-2.1.3 inspection requirements into the daily inspection procedures with the remaining requirements into the periodic inspection procedure. The daily inspections were performed using Procedure 7.6.1, Attachment 3 "Crane Operations Daily Inspection Checklist." Section 1.1 of Attachment 3 required inspection of all functional operating mechanisms for proper operation. This included pushbutton switches, selector switches, master switches, speed controls, and brakes. Step 1.3 required checking for visible leakage on the floor and the crane structure that would indicate leakage from lines, tanks, valves, pumps, and other parts of the hydraulic systems. Step 1.4 required inspecting the hooks for more than 15% in excess of normal throat opening or more than a 10 degrees twist; severe nicks, gouges, or cracks; proper operation of hook latches; damage or

malfunction of hook attachments and securing means; and wear exceeding 10% of original dimensions. The remaining required inspection items from ASME B30.2, Section 2-2.1.3 were inspected periodically per Procedure 7.2.73. Periodic examinations were performed at least annually. Step 4.1.4 required functionally checking the limit switches for the trolley and bridge. Step 4.1.6 required examining and testing the limit switches for the hoist. The hoist ropes were extensively examined per Step 4.2 including looking for improperly applied end connections. The required periodic inspection per this procedure was completed on September 19, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Maintenance Procedure 7.2.73 "Reactor, Turbine, and Auxiliary Turbine Building Crane Examination, Maintenance, and Testing," Revision14, (c) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24

Category: Crane Inspection **Topic:** Crane Operational Testing

Reference: ASME B30.2 (1976) Section 2-2.2.1 Revision 1976

Requirement: Prior to initial use, all new, reinstalled, extensively repaired, or modified cranes shall be tested to insure compliance with this standard including the following functions: (a) hoisting and lowering, (b) trolley travel, (c) bridge travel, (d) limit switches, and (e) locking and safety devices. The trip setting of the hoist devices shall be determined by tests with an empty hook traveling in increasing speeds up to the maximum speed. The actuating mechanism of the limit device shall be located so that it will trip the device under all conditions in sufficient time to prevent contact of the hook or load block with any part of the trolley or crane.

Observation: Cooper's reactor building crane was tested after modifications were made to increase the capacity of the crane from 100 tons to108 tons. The following modifications were performed per Change Evaluation Document (CED) 6028740: replaced the variable frequency drive for the main hoist motor, replaced two lower load cell connection pins with higher strength material pins, increased the size of two six inch welds on the equalizer bar support plate from 1/2 inch to 5/8 inch, installed girder stiffening bars, and replaced the main hoist cable with larger diameter wire ropes. The crane was then tested using Procedure REP-20881-014. The procedure extensively tested the crane including the attributes required by American Society of Mechanical Engineers (ASME) B30.2, Section 2-2.2.1. Section II.C of Procedure REP-20881-014 required testing of the main hoist limit switches (upper, lower, and the redundant power upper/lower switches), the speed of the hoist, the safety devices for the hoist (over-speed detector, overload relays, brake faults, and power faults), and the hoist brakes. Section II.D required testing the trolley travel limit switches in both directions. Section II.E required testing the bridge travel limit switches in both directions. Section II.C, D, and E were performed with no load on the hook, except for one brake test which required a 20 ton load. A 100% load was used to test the overweight limit switch and underweight limit switch in Section III.A, Steps 12 and 13. The functional and load testing procedure was performed and completed on July 15, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Change Evaluation Document (CED) 6028740 "Cooper Nuclear Station Reactor Building Crane Re-Rate," dated June 25, 2010, (c) American

Crane & Equipment Corporation Procedure REP-20881-014 "Site Functional and Load Test Procedure for Reactor Building Crane Controls Upgrade," Revision 0

Category: Crane Inspection **Topic:** Hoist Overload Testing

Reference: APCSB 9-1 (1975) Section B.4.b Issued 1975

Requirement: The complete hoisting machinery should be tested for ability to sustain a load hang-up condition by a test in which the load block attaching points are secured to a fixed anchor or excessive load. The drum should be capable of one full revolution before starting the hoisting test.

Observation: This test was not applicable to the reactor building crane at Cooper. Cooper's crane was equipped with an overweight limit switch that protected the crane by shutting it down if the crane attempted to hoist weight that was greater than expected. This switch was set at 240,000 lbs (120 tons), which was 15% greater than the crane's rated capacity. If the crane's load indicator device determined that the crane was attempting to raise a weight greater than 120 tons, the crane would shut down, protecting itself from any overloading conditions. The overweight limit switch was tested July 15, 2010 during the crane performance test per Procedure REP-20881-014, Section III.A, Step 12.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Crane & Equipment Corporation Procedure REP-20881-014 "Site Functional and Load Test Procedure for Reactor Building Crane Controls Upgrade," Revision 0

Category: Crane Inspection **Topic:** Hoist Two-Block Testing per APCSB 9-1

Reference: APCSB 9-1 (1975) Section B.4.b Issued 1975

Requirement: The complete hoisting machinery should be allowed to two-block during the hoisting test (load block limit and safety devices are bypassed). This test should be conducted without load and at slow speed, to provide assurance of the integrity of the design, equipment, controls, and overload protection devices. The test should demonstrate that the maximum torque that can be developed by the driving system, including the inertia of the rotating parts at the overtorque condition, will be absorbed or controlled prior to two-blocking.

Observation: This test was not performed because the crane was equipped with two upper limit switches that protected the crane from hoisting the load block into itself. NUREG 0554, Section 4.5 "Design Against Two-Blocking" allowed for dual limit switches as an alternative to the mechanical and structural components of the hoisting system having the required strength to resist failure during a two-blocking incident.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975

Category: Crane Inspection **Topic:** Hoist Two-Block Testing per ASME B30.2

Reference: ASME B30.2 (1976) Section 2-3.2.4 Revision 1976

Requirement: Prior to initial use of any hoist during each shift, the operator shall verify operation of the upper limit device under no-load conditions. The block shall be inched into the limit

or run in at slow speed. If the device does not operate properly, the operator shall immediately notify the appointed person.

Observation: Hoist two-block testing was performed prior to the shift when the crane was to be used that day. Procedure 7.6.1, Attachment 3 "Crane Operations Daily Inspection Checklist" was performed each day prior to the shift to check out the crane for proper use. Step 1.2.3 of Attachment 3 required slowly raising the hook to actuate the limit switch.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24

Category: Crane Inspection **Topic:** Hook Inspections - Frequency

Reference: ASME B30.10 (1975) Sections 10-1.4.2 and 10-1.4.6 Revision 1982

Requirement: Hooks shall be inspected monthly during normal service, weekly to monthly during heavy service and daily to weekly during severe service. Hooks shall be inspected for: a) distortion such as bending, twisting or increased throat opening; b) wear; c) cracks, severe nicks, or gouges; d) damaged or malfunctioning latch (if provided); and e) hook attachment and securing means. Hooks having any of the following deficiencies shall be removed from service unless a qualified person approves their continue use and initiates corrective action: a) cracks; b) wear exceeding 10% of the original sectional dimension; c) bend or twist exceeding 10 degrees from the plane of an unbent hook; d) an increase in throat opening of 15% (for hooks without latches); or e) if the latch becomes inoperable.

Observation: The required hook inspections were performed daily before use of the reactor building crane. All the required hook inspection criteria were incorporated into Section 1.4 of Procedure 7.6.1, Attachment 3, “Crane Operator Daily Inspection Checklist”. Procedure Step 1.4.1 required inspection of the hook for having more than 15% in excess of normal throat opening. Step 1.4.2 required inspection for more than a 10% twist. Step 1.4.3 required inspection for severe nicks, gouges, or cracks. Step 1.4.4 required inspection of proper operation of hook latches. Step 1.4.5 required inspection for damage or malfunction of hook attachment and securing means. Step 1.4.6 required inspection of hook wear exceeding 10% of the original dimensions.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.10 “Hooks,” Revision 1982, (b) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24

Category: Crane Inspection **Topic:** Welding

Reference: APCSB 9-1, Sect. B.1.f; ASME B30.2, Sect. 2-1.4.1 Issued 1975

Requirement: All welding on load-sustaining members shall be in accordance with American Welding Society (AWS) structural welding code AWS D1.1, except as modified by AWS D14.1. For low alloy steel the recommendations of Reg Guide 1.50 should be followed.

Observation: During the re-rate work on the crane from 100 tons to 108 tons, the American Welding Society (AWS) code requirements for AWS D1.1 and D14.1 were incorporated into the welding instructions. Attachment 6 "Design Inputs and Requirements" of Change Evaluation Document (CED) 6028740, Section C2.B.2 "Industry Codes and Standards" stated "All work associated with this modification shall be accomplished with applicable

codes, standard specifications..." among which was AWS D14.1. Attachment 8 "Installation/Testing Requirements," Page 8-1 of the CED (Installation Instructions), Sections 1.3.2, 1.3.4 and 1.3.5, with respect to the completed welds, stated “Welds are in accordance with AWS D-1.1 or D-14.”

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (c) Change Evaluation Document (CED) 6028740 "Cooper Nuclear Station Reactor Building Crane Re-Rate," dated June 25, 2010

Category: Crane Inspection **Topic:** Wire Rope Inspection - Annual

Reference: ASME B30.2 (1976) Section 2-2.4.1 (a) Revision 1976

Requirement: An inspection of all ropes shall be made at least annually and a dated report of rope condition kept on file where available to appointed personnel. Sections of rope which are normally hidden during visual and maintenance inspections, such as parts passing over sheaves, should be given close inspection as these are points most likely to fail. Any deterioration resulting in appreciable loss of original strength, such as described below, shall be noted and a determination made as to whether further use of the rope would constitute a hazard: (1) reduction of rope diameter below nominal diameter due to loss of core support, internal or external corrosion or wear of outside wires, (2) a number of broken outside wires and the degree of distribution or concentration of such broken wires, (3) worn outside wires, (4) corroded or broken wires at end connections, (5) corroded, cracked, bent, worn or improperly applied end connection, and (6) kinking, crushing, cutting or unstranding.

Observation: The running ropes of the reactor building crane at Cooper, which were used to lift the spent fuel cask, were inspected annually. A full length wire rope inspection was performed prior to loading each individual cask per Procedure 10.37 or annually per Procedure 7.2.73. Loading Procedure 10.37, Step 3.2.9 required a full length wire rope inspection to have been performed in accordance with Section 4.1 of Procedure 6.MISC.601. The steps of Section 4.1 contained all the required inspection criteria from the American Society of Mechanical Engineers (ASME) B30.2 guidance. Step 4.1.1 required examining the cable throughout its entire length for kinking, crushing, cutting, un-stranding, bird-caging, main strand displacement, core protrusion, or evidence of heat damage. Step 4.1.2 required examining outside wires of the cable. Step 4.13 required examining end connections for corrosion, broken wires, cracking, bending, wear, or improperly applied connections. Step 4.1.4 required examining the cable for broken wires throughout its length including end connections, and then to record the number of broken wires and the locations. Step 4.1.5 required examining the cable for internal or external corrosion throughout its length including end connections and recording discrepancies. The required inspection prior to lifting the first cask per Procedure 6.MISC.601 was completed on April 30, 2010.

For the annual inspection, the steps in Section 4.2 of Procedure 7.2.73 contained all the required inspection criteria. Step 4.2.1.1 required measuring the diameter of the wire ropes to confirm the diameter was greater than the minimum acceptable diameter of 1.1875 inches. Step 4.2.7 required checking for the number of broken outside wires and

determining the distribution or concentration of the broken wires. Step 4.2.8 required examining for worn outside wires. Step 4.2.9 required examining the ropes for corroded or broken wires at end connections. Step 4.2.10 required checking for severely corroded, damaged, bent, worn, or improperly applied end connections. Step 4.2.11 required checking for kinking, crushing, cutting, or un-stranding. The required annual inspection per this procedure was completed on September 19, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Nuclear Performance Procedure 10.37 "Dry Shield Canister Loading," Revision 0, (c) Surveillance Procedure 6.MISC.601 "Reactor Building Crane Inspection or Lift and Hold Operability Test for Cask Handling Operations," Revision 10, (d) Maintenance Procedure 7.2.73 "Reactor, Turbine, and Auxiliary Turbine, Building Crane Examination, Maintenance, and Testing," Revision 14

Category: Crane Inspection **Topic:** Wire Rope Inspection - Daily Checks

Reference: ASME B30.2 (1976) Section 2-2.4.1 (a) Revision 1976

Requirement: All running ropes in continuous service shall be visually inspected once each working day. Any deterioration resulting in appreciable loss of original strength, such as described below, shall be noted and a determination made as to whether further use of the rope would constitute a hazard: (1) reduction of rope diameter below nominal diameter due to loss of core support, internal or external corrosion or wear of outside wires, (2) a number of broken outside wires and the degree of distribution or concentration of such broken wires, (3) worn outside wires, (4) corroded or broken wires at end connections, (5) corroded, cracked, bent, worn or improperly applied end connection, and (6) kinking, crushing, cutting or unstranding.

Observation: The running ropes of the reactor building crane at Cooper, which were used to lift the spent fuel cask, were inspected prior to use each day. The steps of Procedure 7.6.1, Attachment 3 "Crane Operator Daily Inspection Checklist," Section 1.5 contained all the required inspection criteria for wire ropes as specified in the American Society of Mechanical Engineers (ASME) B30.2 guidance. The reactor building crane main hoist consisted of two wire ropes of 1 ¼ inch diameter and a length of 855 feet each. Step 1.5.1 required inspection for kinking, crushing, cutting, or un-stranding and bird-caging, main strand displacement, or core protrusion. Step 1.5.2 required inspection for reduction of rope diameter below nominal due to loss of core support, internal or external corrosion, or wear of outside wires. Step 1.5.3 required inspection for the number of broken outside wires and the degree of distribution or concentration of such broken wires. Step 1.5.4 required inspection for worn outside wires. Step 1.5.5 required inspection for corroded or broken wires at end connections. Step 1.5.6 required inspection for corroded, cracked, bent, worn, or improperly applied end connections.

Cooper was following ASME B30.2, Revision 1983 for guidance on the frequency of full length wire rope inspections. The 1976 revision does not specify if the daily wire rope inspection should be the full length or not. However the 1983 revision does, stating that the frequency shall be determined by a qualified person. At Cooper, the full length wire rope inspection was performed prior to loading of each individual cask. Loading Procedure 10.37, Step 3.2.9 required a full length wire rope inspection to be performed in accordance with Section 4.1 "Cable Inspection" of Procedure 6.MISC.601. The steps

in Section 4.1 contained all the requirements of ASME B30.2. Step 4.1.1 required examining the cable throughout its entire length for kinking, crushing, cutting, un-stranding, bird-caging, main strand displacement, core protrusion, or evidence of heat damage. Step 4.1.2 required examining outside wires of the cable. Step 4.13 required examining end connections for corrosion, broken wires, cracking, bending, wear, or improperly applied connections. Step 4.1.4 required examining the cable for broken wires throughout its length including end connections, and then to record the number of broken wire and the locations. Step 4.1.5 required examining the cable for internal or external corrosion throughout its length including end connections and recording discrepancies. The required inspection prior to lifting the first cask per Procedure 6.MISC.601 was completed on April 30, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24, (c) Nuclear Performance Procedure 10.37 "Dry Shield Canister Loading," Revision 0, (d) Surveillance Procedure 6.MISC.601 "Reactor Building Crane Inspection or Lift and Hold Operability Test for Cask Handling Operations," Revision 10

Category: Crane Inspection **Topic:** Wire Rope Replacement Criteria

Reference: ASME B30.2 (1976) Section 2-2.4.2 (b) Revision 1976

Requirement: Conditions such as the following should be sufficient reason for questioning continued use of the rope or increasing the frequency of inspection: (1) twelve randomly distributed broken wires in one rope lay or four broken wires in one strand of one rope lay (see Errata sheet); (2) wear of one-third of the original diameter of outside individual wires; (3) kinking, crushing, bird caging or any other damage resulting in distortion of the rope structure; (4) evidence of heat damage; and (5) reduction from nominal diameter of more than 3/32 inch for wire ropes with a diameter of 1-1/4 inch

Observation: The replacement criteria per American Society of Mechanical Engineers (ASME) B30.2, for wire ropes, was included in Procedure 7.2.73. The reactor building crane main hoist consisted of two wire ropes of 1-1/4 inch (1.250 inch) diameter and a length of 855 feet each. Procedure 7.2.73, Step 4.2.1.1 required the main hoist wire rope minimum acceptable diameter to be greater than 1-3/16 inch (1.1875 inch). This was an acceptable practice as the ASME B30.2, Section 2-2.4.2 "Rope Replacement" allowed a reduction in nominal diameter of up to 3/32 inch compared to Procedure 7.2.73 of 1/16 inch (i.e. 2/32 inch). Step 4.2.2.2 required replacement of the wire rope if it was below the minimum acceptable diameter. Step 4.2.7 required checking for the number of broken outside wires and to determine the distribution or concentration of the broken wires. Twelve randomly distributed broken wires in one lay or four broken wires in one strand of one lay was unacceptable and required replacement. Step 4.2.8 required inspection for worn outside wires. Step 4.2.11 required checking for kinking, crushing, cutting, or un-stranding. Step 4.2.12 required inspection for evidence of heat damage. The required annual inspection per this procedure was completed on September 19, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Maintenance Procedure 7.2.73 "Reactor, Turbine, and Auxiliary Turbine, Building Crane Examination, Maintenance, and Testing," Revision 14

Category: Crane Licensing Basis **Topic:** Crane 70 Ton Limit

Reference: License DPR-46, Tech Spec T 3.9.2 10/01/08

Requirement: The original Technical Specification T.3.9.2 Limiting Condition of Operation stated "The spent fuel cask shall weigh less than or equal to 140,000 lbs (70 tons) and the fuel cask handling equipment used shall be operable in the restricted mode." This restriction was removed in order to perform cask loading operations using the NUHOMS cask system.

Observation: Cooper Nuclear Station provided an adequate basis for removing the spent fuel cask weight restriction of 70 tons from Technical Requirement T.3.9.2 in Licensing Basis Document Change Request (LBDCR) 2010-023. The reactor building crane was originally described as having a rating of 100 tons in the Amendment 33 submittal of Cooper's Updated Safety Analysis Report (USAR). The 100 ton rating was based on guidance from the Crane Manufacturers Association of America (CMAA) Guide #70, American Society of Mechanical Engineers (ASME) B30.2, and Occupational Safety and Health Administration (OSHA) requirements. Cooper's crane was designed and procured in the late 1960's prior to the development of the Atomic Energy Commission's (now NRC) Branch Technical Position APCSB 9-1, NUREG 0612, and NUREG 0554 and as such did not originally meet the full intent of these subsequent document. During review of the crane for License Amendment 35, the NRC staff identified three issues in non-conformance with APCSB 9-1. These included: (1) lack of a redundant limit switch to prevent two-blocking with a power disconnect, (2) insufficient margin in the lead line portion of the wire rope, and (3) too large of a fleet angle in the reeving. These issues were addressed by the licensee by a commitment to modify the crane to include the specified two-blocking protection, limit the maximum cask weight to 70 tons, and add a surveillance test requirement to inspect the wire rope and replace it if specified criteria were not satisfied. The 70 ton load limit placed in the licensee's Technical Specifications T.3.9.2 along with the wire rope inspection and replacement program provided an equivalent level of protection to assure that accelerated wire rope wear would be detected well before a problem could occur and satisfied the NRC's concerns in addressing the non-conformances. These issues were documented in the NRC's Safety Evaluation Report (SER) entitled "Supporting Approval of Facility Modifications to Reduce the Probability of a Fuel Cask Drop Accident to an Acceptably Low Level and Amendment No. 35 to License No. DPR-46." The purpose of the 70 ton restriction was to ensure that if heavier casks were used, a reanalysis would be required. Page 2 of the SER stated that "If larger casks are used, additional analysis will be required to assure safety margins are maintained."

With the License Amendment 178 conversion to Improved Standard Technical Specifications in July 1998, the Spent Fuel Cask Movement Technical Specification was relocated to the Technical Requirements Manual by the licensee. Changes to requirements in the Technical Requirements Manual were subject to the provisions of 10 CFR 50.59. The licensee made numerous crane modifications and established a new analysis bases for demonstrating the acceptability of lifting the 108-ton Transnuclear OS197H transfer cask, while preserving the crane licensing basis that the probability of a cask drop remained acceptably low. The licensee concluded that the changes and bases collectively provided the design and licensing bases changes needed to remove the 70-ton restriction in TLCO 3.9.2. A 50.59 screening was completed. To address the

concerns of the 70 ton restriction, the wire rope was replaced with a rope with a higher yield and breaking strength and the crane was modified to include two-blocking protection. In 1979, the NRC endorsed the use of NUREG 0554 for single failure proof cranes. APCSB 9-1, Section 3. f. required the drum to lead sheave in the block to not exceed 3.5 degrees and the fleet angles for the rope between individual sheaves to not exceed 1.5 degrees. NUREG 0554, Section 4.1 required both the drum to lead sheave angle and the angles between individual sheaves to not exceed 3.5 degrees at any one point during hoisting except that for the last three feet of maximum lift elevation, the fleet angle may increase slightly. Report REP-20881-001, Section 4.1 "Reeving System" stated that the hoist system met the NUREG 0554 requirement of 3.5 degrees, except for the last 3 feet of maximum lift elevation. The fleet angles were documented to be 3.58 degrees at the maximum lift height for the angle from the drum to the lead block. Angles between individual sheaves ranged from 0.35 to 2.87 degrees. Cooper's reactor building crane thus meets the NUREG 0554 requirements in regards to fleet angles. Further, the Cooper crane met the newer industry guidance specified in Section 5426.1 of ASME NOG-1-2004. This guidance established a limit of 3.5 degrees for the fleet angle to the drum with a limit of 4 degrees for the last three feet at maximum lift height, and a sheave fleet angle limit of 3.5 degrees with a limit of 4.5 degrees for the last 3 feet of maximum lift height.

Documents Reviewed: (a) Licensing Basis Document Change Request (LBDCR) 2010-023 "Revise TLCO to Delete the 140,000 lbs Spent Fuel Cask Weight Restriction," Revision 0, (b) Procedure 08, Attachment 3 "50.59 Screen Form," for Activity LBDCR 2010-023 "Revise TLCO to Delete the 140,000 lbs Spent Fuel Cask Weight Restriction," dated August 27, 2010, (c) Letter from Dennis Ziemann, NRC to J. M. Pilant, Nebraska Public Power District entitled "Request for Additional Information Related to Plans and Analysis for Use of a Modified Overhead Crane Handling System," dated October 16, 1975, (d) Cooper White Paper "Justification for TRM Change to Remove Spent Fuel Cask Movement Weight Restriction", Draft, (e) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979, (f) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (g) NRC Safety Evaluation Report "Supporting Approval of Facility Modifications to Reduce the Probability of a Fuel Cask Drop Accident to an Acceptably Low Level and Amendment No. 35 to License No. DPR-46" dated February 28, 1977, (h) American Crane & Equipment Report REP-20881-001, “NUREG 0554/0612 Compliance/Safety Analysis Report” dated September 27, 2007, (i) American Society of Mechanical Engineers (ASME) NOG-1-2004 "Rules for Construction of Overhead and Gantry Cranes", date May 16, 2005, (j) Crane Manufacturers Association of America (CMAA) Guide #70 "Top Running and Gantry Type Multiple Girder Electric Overhead Traveling Cranes," released 1971, (k) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (l) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24

Category: Crane Licensing Basis **Topic:** Crane Support Structure

Reference: Condition Report CR-CNS-2008-04810 6/19/08

Requirement: The reactor building crane was limited to 70 tons due to the inadequacy of calculations performed for the building support structure seismic response.

Observation: The issue identified in Condition Report CR-CNS-2008-04810 related to the seismic calculations for the reactor building were resolved. The issue focused on the omission of a safe shutdown earthquake (SSE) calculation for the dynamic load on the original crane and related to the adequacy of the crane and building supports for frequencies lower than 33 hertz (Hz) at Elevation 1047 feet in the reactor building. Cooper Nuclear Station had contracted with Stevenson and Associates to develop a modification package for the reactor building structure as part of the crane up-rate from 100 tons to 108 tons. Stevenson and Associates developed a model to determine where modifications would be needed. It was concluded that minimal modifications would be needed such as replacement of the clips which tied the crane girders to the structural steel of the building, and the tightening of bolts. Burns and Roe, who performed the original calculations for the reactor building structure, was contracted to perform a "peer check" of Stevenson and Associates calculations because the licensee had expected more extensive modifications to be required. Burns and Roe completed an independent analysis (without seeing the conclusions of Stevenson and Associates) and concluded that no modifications were necessary. The models used by the two firms had several differences. First, Stevenson and Associates used the ultimate strength of the structural steel which resulted in the plastic deformation of the steel. Additionally, Stevenson and Associates used a zero period acceleration of 33 Hz which was not consistent with the licensee's specific seismic response spectra. Stevenson and Associates also did not include the crane steel modifications in the original model. Burns and Roe, however, used the yield strength of the structural steel in their model which did not result in the plastic deformation of the steel. Burns and Roe also used the site specific zero period acceleration, as opposed to 33 Hz, and included the crane steel modifications in their model.

The primary contention in Condition Report 2008-04810 was that the reactor building superstructure cannot be considered dynamically rigid in the seismic analyses because Calculation NEDC 07-077 demonstrated that the natural frequency of the crane rail girder in the weak axis was 20.8 Hz. Region IV inspectors, with consultation from the NRC's Office of Nuclear Reactor Regulations, determined that even though this natural frequency was less than 33 Hz, it was still in the rigid range for Cooper Nuclear Station. The Cooper Nuclear Station seismic response spectra curves for the reactor building superstructure reached zero period acceleration values at approximately 20 Hz. Structures with natural frequencies in excess of this value can be expected to exhibit an in-phase, pseudo-static response.

Upon review of the Burns and Roe model and conclusions, the licensee identified additional safety margin, in that, a large margin was added to the calculated structural steel load used in the original model. Stevenson and Associates ultimately included the crane steel modifications in their model, and reached the same conclusion as Burns and Roe; that no additional modifications were necessary to support the increased suspended load coincident with a design basis earthquake. NRC inspectors independently verified that the final Stevenson and Associates conclusion, which was documented in Calculation NEDC 10-036, supported the conclusion that no additional modifications to the reactor building structure were necessary. Calculation NEDC 10-036 created a 3-D finite element model of the reactor building's superstructure including the crane support structure, crane rail, and crane bridge. The analysis was performed for a 108 ton rating

for the crane. Engineering Evaluation 10-024, Section 4.2.2 "Whiting Crane" stated that the calculations also demonstrated that the analysis that bounded the 108 ton up-rated load also included the capability for conducting the occasional overload condition such as during the 125% load test.

Documents Reviewed: (a) Condition Report CR-CNS-2008-04810 "CNS Reactor Building Crane Seismic Upgrade for ISFSI Load," initiated June 19, 2008, (b) Condition Report CR-CNS-2009-02495 "Reactor Building Crane Seismic Analysis," initiated March 26, 2009, (c) Engineering Design Calculation (NEDC) 07-077 "Seismic Evaluation of Reactor Building with Loaded Reactor Building Crane," Revision 0, (d) Engineering Evaluation 10-024 "Reactor Building Superstructure and Crane," Revision 0, (e) Engineering Design Calculation (NEDC) 10-036 "Reactor Building Superstructure Evaluations," Revision 31

Category: Crane Licensing Basis **Topic:** Crane Up-Rating from 100 Tons to 108 Tons

Reference: Change Evaluation Document (CED) 6028740 July 7, 2010

Requirement: An upgrade to structural and load bearing components is needed to increase the rated capacity of the reactor building crane from 100 tons to 108 tons in preparation for lifting a loaded NUHOMS transfer cask.

Observation: Upgrades to the structural and load bearing components of the current crane were completed by the licensee in Change Evaluation Document (CED) 6028740 to up-rate the crane capacity from 100 tons to 108 tons. None of these changes required modifications to the reactor building structure. Changes that were made included replacement of the variable frequency drive for the main hoist motor, replacement of two lower load cell connection pins, increasing the size of two welds on the equalizer bar support plate, enlarging the end holds for the rope anchors in the two vertical end plates of the equalizer assembly, installing girder stiffening A-36 bars and stitch welding two flat bars approximately 60 feet on top of girder A, placing and stitch welding a flat bar approximately 60 feet on top of girder B, installing trolley parking lock rack splice reinforcement bars to girder B, and replacing the main hoist cables.

Documents Reviewed: (a) Change Evaluation Document (CED) 6028740 "Re-Rate of Reactor Building Crane Hoist (O-HST-020) to 108 Tons," dated July 7, 2010, (b) Change Evaluation Document (CED) 6023100 "Reactor Building Crane Upgrade," dated September 27, 2007, (c) Engineering Design Calculation (NEDC) 07-077 "Seismic Evaluation of Reactor Building with Loaded Reactor Building Crane Calculation," Revision 0, Status 2, dated July 13, 2010, (d) Engineering Evaluation 10-024 "Reactor Building Superstructure and Crane," Revision 0

Category: Crane Licensing Basis **Topic:** Generic Issue 199 - Seismic

Reference: NRC Information Notice 2010-18 Issued 2010

Requirement: The NRC has updated the seismic hazard models for nuclear facilities in the central and eastern U. S. due to the New Madrid Fault and other regional faults. This new data affects the Cooper site. New seismic hazards analysis will become available in early 2011.

Observation: The licensee has been following this issue for several years and had participated in meetings related to the new studies that have been completed on the increased seismic

hazard estimates discussed in Generic Issue 199. The licensee had evaluated the seismic studies on the effect at the Cooper Nuclear Station and believed that the impact on the plant will not be significant once the NRC issued their final seismic hazards estimates.

Documents Reviewed: (a) NRC Information Notice 2010-18 "Generic Issue 199 Implications of Updated Probabilistic Seismic Hazards Estimates in Central and Eastern United States on Existing Plants [NRC ADAMS Accession No. ML101970221], dated September 2, 2010

Category: Crane Load Testing **Topic:** Cold Proof Testing

Reference: NUREG 0554, Section 2.4; NUREG 0612, C-2 (2) 1979/1980

Requirement: Minimum operating temperatures for the crane should be specified to reduce the possibility of brittle fracture of the ferritic load-carrying members of the crane. The minimum temperature can be determined by: 1) a drop weight test per ASTM E-208, 2) a Charpy test per ASTM A-370, or 3) a 125% cold proof test. If the crane is made of low alloy steel such as ASTM A514, cold proof testing should be done. If cold proof testing is omitted, the default minimum crane operating temperature is 70 degrees F. For crane operation at temperatures below 70 degrees F, cold proof testing must be performed and the ambient temperature at which the testing is conducted becomes the minimum crane operating temperature.

Observation: The licensee did not document the temperature of the 125 percent load test performed in 1976 because the building was climate controlled and heavy lifts were only allowed for temperatures greater than or equal to 70 degrees Fahrenheit. Procedures 10.37, 10.37.1, 10.38, and 10.38.1 limited the reactor building crane operation to an ambient temperature of greater than or equal to 70 degrees Fahrenheit at the crane girders. This limitation was specified in the Precautions and Limitations Section of the procedures.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 0, (c) Nuclear Performance Procedure 10.37.1 "Shielded Canister Unloading," Revision 0, (d) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (e) Nuclear Performance Procedure 10.38.1 "Dry Shielded Canister Unsealing," Revision 1, (f) Maintenance Procedure 7.2.73 "Reactor, Turbine, and Auxiliary Turbine Building Crane Examination, Maintenance, and Testing," Revision 14

Category: Crane Load Testing **Topic:** Load Testing - Dynamic 100% Load

Reference: APCSB 9-1 (1975) Section B.4.b Issued 1975

Requirement: After the 125% static load test, the crane should be given a full performance test with 100% of the maximum critical load attached, for all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices. The crane handling system should demonstrate the ability to lower and move the design rated load by manual operation and with the use of emergency operating controls and devices which have been included in the handling system.

Observation: All required performance testing with 100% of the rated load was completed prior to fuel loading activities. The 100% load test was performed using Procedure REP-20881-014 and was completed on July 15, 2010. The weight used to perform the load test was 218,200 lbs (109.1 tons). This slightly exceeded the load rating of the crane at 108 tons

and the actual weight of a loaded cask that would be lifted during cask loading operations. The calculated maximum weight of the spent fuel transfer cask, fully loaded with spent fuel, full of water, and with the shield plug installed was 212,356 lbs (106.2 tons) per Calculation Number 08-042. All limiting and safety control devices were tested as required in Section II of Procedure REP-20881-014 with no load on the hook. The full range of motion of the crane with the 109.1 ton test load was then performed in accordance with Section III of Procedure REP-20881-014. This testing included raising and lowering the test load at various speeds and moving the trolley and bridge through a wide range of motions at various speeds. Additionally, manually lowering of the load (109.1 tons) was completed as required by Section III.A, Steps 14 and 15. This was completed by blocking the south hoist brake open and manually lowering the test load by pulling up on the brake release handle (Step 14). The opposite side was then tested by blocking the north hoist brake open and manually lowering the test load by pulling on the brake release handle (Step 15). No discrepancies were noted during the load test.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Crane & Equipment Corporation Procedure REP-20881-014 "Site Functional and Load Test Procedure for Reactor Building Crane Controls Upgrade," Revision 0, (c) Calculation Number 08-042 "Transnuclear Transfer Cask (TC) and Dry Shielded Canister (DSC) Weights for Various Spent Nuclear Fuel Loading Configurations," Revision 0

Category: Crane Load Testing **Topic:** Load Testing - Static 125% Load

Reference: APCSB 9-1 (1975) Section B.4.b Issued 1975

Requirement: The crane should be static load tested at 125% of the maximum critical load. The test should include all positions of hoisting, lowering, and trolley and bridge travel as recommended by the designer and manufacturer.

Observation: All required performance testing with 125% of the rated load was completed prior to fuel loading activities. The 125% load test was performed using Procedure REP-20881-014 and was completed on July 21, 2010. The weight used to perform the test was 266,000 lbs (133 tons). This was 123.1% of the load rating of the crane (108 tons). American Society of Mechanical Engineers (ASME) B30.2, Section 2-2.2.2 "Rated Load Test" stated that the test load shall not be more than 125% of the rated load and shall consist of testing the brakes and moving the load the full length of the bridge and trolley. The 125% rated load test was performed at Cooper using Section IV of Procedure REP-20881-014. Section IV included raising and holding the 125% load to verify the brakes would hold the load, de-energizing the main contact during a lowering test to verify the brakes would engage and prevent a drop of the load, raising and lowering the load at varying speeds, operating the bridge and trolley in all directions over the longest distances possible while varying the speed, de-energizing the main contact and confirming that the bridge and trolley brakes engaged and stopped the movement.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) American Crane & Equipment Corporation Procedure REP-20881-014 "Site Functional and Load Test Procedure for Reactor Building Crane Controls Upgrade," Revision 0, (c) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976

Category: <u>Crane Load Testing</u> **Topic:** <u>Maximum Weight of Load</u>

Reference: UFSAR 1004, Section K.3.4.3 Revision 10

Requirement: The maximum weight of the transfer cask containing the canister filled with water and fuel (including dynamic loads) that will be lifted by the crane is to be verified to be within the crane's rated capacity.

Observation: The maximum weight of the transfer cask containing the canister filled with water and spent fuel that will be lifted by the crane was within the crane's rated capacity. The crane was rated for a 108 ton load on the hook. Calculation 08-042 concluded that the maximum cask weight was about 106.2 tons when lifting the heaviest load which would be the transfer cask containing the loaded canister coming out of the spent fuel pool with the canister filled with water and the canister shield lid in place. The weight calculations included the weight of the canister shell assembly, canister top shield plug and covers, canister internal basket, 61 spent fuel assemblies, the weight of the water inside the canister, the transfer cask, the transfer cask shield water, and the transfer cask/canister annulus water. The dynamic loads were accounted for in the dynamic analysis and were not included in the 106.2 static weight determination.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Calculation 08-042 "Transnuclear Transfer Cask (TC) and Dry Shielded Canister (DSC) Weights for Various Spent Fuel Loading Configurations," Revision 0

Category: <u>Crane Operations</u> **Topic:** <u>Brake Test Prior to Lift</u>

Reference: ASME B30.2 (1976), Section 2-3.2.3 (g) Revision 1976

Requirement: The operator shall check the hoist brakes at least once each shift if a load approaching the rated load is to be handled. This shall be done by lifting the load a short distance and applying the brakes.

Observation: The brake test was performed as required before each lift of the transfer cask. All procedures utilizing the transfer cask contained the required brake test. The lifts performed at Cooper that involved the spent fuel canister were governed by Procedures 10.37, 10.38, and 10.39. Prior to insertion of the transfer cask into the spent fuel pool, Procedure 10.37, Step 7.11 required slowly lifting the transfer cask high enough to check the bottom of the transfer cask for foreign material. This action involved applying the brakes and holding the canister. Prior to raising the transfer cask out of the spent fuel pool, Procedure 10.38, Steps 4.27 and 4.28 required slowly lifting the transfer cask a few inches and applying the brakes to verify the brakes did not slip. Prior to moving the transfer cask out of the cask wash down area to the transport trailer, Procedure 10.39, Steps 6.10 and 6.11 required lifting the transfer cask approximately 6 inches and holding to verify the brakes did not slip.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 0, (c) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (d) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 8

Category: Crane Operations **Topic:** Commitment to Inspect Wire Rope

Reference: NRC Safety Evaluation Report #35 February 28, 1977

Requirement: NRC Safety Evaluation Report #35 stated: "The licensee, by letter dated April 6, 1976, has committed to incorporate into the CNS Technical Specifications, a specific program of wire rope inspection and replacement, the purpose of which would be to ensure the wire rope will be maintained as close as practicable to original design safety factors at all times. This inspection and replacement program provides an equivalent level of protection to the methods suggested in our wire rope safety and crane fleet angle criteria and will assure that accelerated wire rope wear will be detected before crane use and satisfies our concerns; and on this basis, we conclude that the crane reeving system is acceptable.

Observation: The commitment to perform a wire rope inspection prior to fuel cask handling operations was required by the licensee's Technical Requirements Manual (TRM). Surveillance TSR 3.9.2.1 required inspection of the rope, hooks, slings, shackles, and other operating mechanisms prior to fuel cask handling operations. The requirement specified to replace the wire rope, if any of the following conditions existed: a) twelve randomly distributed broken wires in one lay or four broken wires in one strand of one rope lay, b) wear of one-third the original diameter of outside individual wires, c) kinking, crushing, or any other damage resulting in distortion of the rope, d) evidence of any type of heat damage, e) reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8 inch to 1-1/4 inch inclusive. The wire rope inspection was required in Procedure 10.37, Step 3.2.9 prior to lifting a loaded cask. Step 3.2.9 stated, "Ensure a full length wire rope inspection has been performed in accordance with Section 4.1 "Cable Inspection of Procedure 6.MISC.601 (TSR 3.9.2.1)."

Documents Reviewed: (a) NRC Safety Evaluation Report #35 "Supporting Approval of Facility Modifications to Reduce the Probability of a Fuel Cask Drop Accident to an Acceptably Low Level and Amendment No. 35 to License No. DPR-46," dated February 28, 1977, (b) Cooper's Technical Requirements Manual, dated August 30, 2010, (c) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 0

Category: Crane Operations **Topic:** Crane Maintenance Program

Reference: USAR Section 4.4.2 May 30, 2000

Requirement: A preventive maintenance program for the reactor building crane has been established based upon the crane manufacturer's recommendations.

Observation: The Whiting crane manufacturer's recommended preventative maintenance program was incorporated into the Cooper maintenance program. Cooper’s reactor building crane was classified as a Class A “Standby or Infrequent Service” crane. The Whiting Corp. vender Manual VM-0176 was reviewed by the NRC inspectors and verified to have been properly incorporated into Cooper’s crane procedures. Both the “General Inspection & Maintenance Schedule” and “Lubrication Schedule” chart attributes from manual VM-0176 were incorporated into Procedure 7.2.73 for annual/periodic maintenance and inspection, Procedure 6.Misc.601 for periodic inspection, Procedure 7.6.1 for daily use inspections, and Cooper’s Work Order/Maintenance Order Database for crane lubrication.

Documents Reviewed: (a) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24, (b) Cooper Nuclear Station Vendor Manual VM-0176 "Whiting Corp 100/5 Ton Reactor Building Crane," Revision 40, (c) Maintenance Procedure 7.2.73 "Reactor, Turbine, and Auxiliary Turbine Building Crane Examination, Maintenance, and Testing," Revision14, (d) Surveillance Procedure 6.MISC.601 "Reactor Building Crane Inspection or Lift and Hold Operability Test for Cask Handling Operations," Revision 10, (e) Maintenance Procedure 7.6.1 "Reactor Building Crane Operation," Revision 24

Category: Crane Operations **Topic:** Height Limit During Cask Movement

Reference: N/A

Requirement: For single failure proof cranes, the cask height during movement should be sufficiently high to allow for engaging of the brakes during an uncontrolled descent before the load would impact the floor.

Observation: Procedure 10.38, Step 2.16 established a minimum height above the floor for the transfer cask during movement on the refueling floor of 6 to 16 inches. During the 2010 load test of the reactor building crane, a brake test was performed in which the 100% load was raised approximately 3 feet. The load was then lowered at maximum speed until approximately one foot off the floor, at which time the main contact on the crane was de-energized. The crane stopped the load within 1/2 inch. This distance was well within the recommendation of NUREG 0554, Section 6.1 which specified that the load should stop within a maximum of three inches. As such, the minimum height limits in Procedure 10.38 provided ample distance for the crane brakes to engage during an uncontrolled descent.

Documents Reviewed: (a) Nuclear Performance Procedure 10.38, "Dry Shielded Canister Sealing," Revision 4, (b) American Crane & Equipment Corporation Procedure REP-20881-014 "Site Functional and Load Test Procedure for Reactor Building Crane Controls Upgrade," Revision 0, (c) NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants," published May 1979

Category: Crane Operations **Topic:** Minimum of Two Wraps of Rope

Reference: ASME B30.2 (1976), Section 2-3.2.3 (h) Revision 1976

Requirement: The load shall not be lowered below the point where less than two wraps of rope remain on each anchorage of the hoisting drum unless a lower-limit device is provided, in which case no less than one wrap shall remain.

Observation: When the spent fuel transfer cask was down-ended onto the tractor trailer at the lowest level of the reactor building, both the left and right lays of rope contained at least a dozen wraps of wire rope around their respective drums. The reactor building crane utilized two wire ropes to lower and raise the crane's hook. The inspector noted during the dry run demonstration that the length of the crane's wire ropes were sufficient to perform the required heavy lift.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976

Category: Crane Operations **Topic:** Minimum Operating Temperature

Reference: APCSB 9-1 (1975), Section B.1.b Issued 1975

Requirement: The maximum and minimum temperature for operations should be specified. Fracture toughness for the steel structural materials should be considered. Plate thickness, with a margin for the lowest operating temperatures, should determine the type of steel that can be used with or without toughness tests.

Observation: Procedure 10.39, Step 2.9 stated "Reactor building crane use is limited to ambient temperature of equal to or greater than 70 degrees F at the crane girders." The 70 degrees F limit was an acceptable minimum temperature limit as stated in NUREG 0612, Page C-2, Step 2. No maximum temperature limit was established since the temperature in the reactor building does not reach high temperatures that could affect crane operations. On April 6, 1976, Nebraska Public Power District responded to an NRC request for additional information on the crane. Their response to Question 1.b related to cold proof testing and stated "Cold proof testing of the crane at 125% of the design rated load has already been accomplished at Cooper. Temperature of the crane at the time of the test was in excess of 50 degrees F. Immediately after completion of the 125% cold proof test, all major load bearing welds were visually inspected and weld gauge sizes were used to check the weld size throughout the crane structure. In addition to the above, all load bearing welds on the new trolley were magnetic particle tested in the shop as part of the fabrication of the trolley." Since Cooper did not document the actual temperature of the cold proof test, the NUREG 0612 limit of 70 degrees F was being used as the minimum temperature limit in the current Cooper procedures.

Documents Reviewed: (a) NRC Branch Technical Position APCSB 9-1 "Overhead Handling Systems for Nuclear Power Plants," issued 1975, (b) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 8, (c) Letter (CNSR766070) from J. M. Pilant, Nebraska Public Power District to Dennis Ziemann, NRC, entitled "NRC Request for Additional Information Cooper Nuclear Station Redundant Crane NRC Docket No. 50-298, DPR-46," dated April 6, 1976, (d) NUREG 0612 "Control of Heavy Loads at Nuclear Power Plants," issued July 1980

Category: Crane Operations **Topic:** Qualification For Crane Operator

Reference: ASME B30.2 (1976), Sections 2-3.1.2 Revision 1976

Requirement: Crane operators shall be required to pass a written or oral examination and a practical operating examination specific to the type of crane to be operated. In addition, the operator shall: (1) have vision of at least 20/30 Snellen in one eye and 20/50 in the other with or without corrective lenses; (2) be able to distinguish colors regardless of their position; (3) have sufficient hearing capability for the specific operation with or without hearing aids; (4) have sufficient strength, endurance, agility, coordination and reaction speed for the specific operation; (5) not have physical defects or emotional instability which could render the operator a hazard to himself or others or could interfere with the operator's safe performance of the crane; (6) not be subject to seizures, loss of control or dizziness; and (7) have normal depth perception and field of vision.

Observation: The crane operators were tested to and met the requirements of the American Society of Mechanical Engineers (ASME) B30.2, Section 2-3.1.2 guidance. Procedure 7.1.10, Section 4.1 included the same physical requirements as ASME B30.2 for crane

operators. Step 4.1.1 required vision of at least 20/30 Snellen in one eye and 20/50 in the other. Step 4.1.2 required operators to be able to distinguish colors regardless of positions. Step 4.1.3 required hearing adequate for a specific operation. Step 4.1.4 required operators to have sufficient strength, endurance, agility, coordination, and speed or reaction to meet the demand of equipment operation. Step 4.1.5 required the individuals to not have physical defects or emotional instability which could render the operator a hazard to themselves or others. Step 4.1.6 required operators to not have evidence of seizures or loss of physical control. Step 4.2 required operators to have normal depth perception, field of vision, reaction time, manual dexterity, coordination, and no tendencies of dizziness. The crane operators designated to perform crane operations with the transfer cask met the ASME B30.2 requirements. The two crane operators completed the physical exams on August 30, 2010. Additionally, Procedure 7.1.10, Step 4.4.1 required crane operators to be trained and take a crane written/oral exam. Step 4.4.2 required all crane operators to take a practical exam. A score of greater than 80% was required to pass the test. Both crane operators had completed the required training to perform ISFSI crane operations and had passed all the required exams. The exams were taken on August 11, 2010.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes," Revision 1976, (b) Maintenance Procedure 7.1.10 "Qualification for Crane or Hoist Operators and Riggers," Revision 4

Category: <u>Drying/Helium Backfill</u> **Topic:** <u>Helium Backfill Final Pressure</u>

Reference: CoC 1004, Tech Spec 1.2.3.a Amendment 9

Requirement: The 61BTcanisters are backfilled with helium to a pressure of 1.5 to 3.5 psig. The pressure must remain stable for 30 minutes after filling.

Observation: The first canister loaded was backfilled with helium to a pressure of 2.5 psig and remained stable for the required 30 minutes, with a final helium pressure of 2.443 psig. This met the requirement from Technical Specification 1.2.3.a. Procedure 10.38, Section 11 "Final DSC Helium Backfill" provided for the final helium backfilling of the canister to meet the technical specification requirements. Step 11.2 stated "Ensure 99.995% purity helium supply is connected to HE-1, helium inlet valve." Step 11.7 stated "Using the helium inlet valve HE-1, pressurize DSC (canister) cavity to between 1.6 and 3.4 psig as indicated on compound pressure gauge PI-3. Step 11.9 stated "Continuously monitor compound pressure gauge PI-3 to verify DSC (canister) cavity pressure is stable for greater than or equal to 30 minutes between 1.6 and 3.4 psig. Steps 11.9.1 and 11.9.2 recorded the helium pressure reading at the start of the 30 minutes and at the end. Step 11.10 was the sign-off by the cask loading supervisor to confirm that the Technical Specification 1.2.3.a requirement had been met.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4

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Category: Drying/Helium Backfill **Topic:** Vacuum Drying Final Pressure

Reference: CoC 1004, Tech Spec 1.2.2 Amendment 9

Requirement: All canisters must be vacuum dried to 3 mm Hg (torr) or less and held for 30 minutes or more. This level of dryness must be achieved in both the initial pump-down and the final pump-down.

Observation: The first canister was dried to less than 3 torr and held below that level for 30 minutes. Procedure 10.38, Section 9.0 "Initial Vacuum Drying" provided instructions for performing the initial vacuum drying of the canister. Section 10 "Initial Helium Backfill" provided instructions for filling the canister with helium and performing the second and final vacuum drying. After the inner lid was welded in-place and all water was drained from the canister, initial vacuum drying was performed in a step process which dried the canister to several pre-selected levels. At each level, the line to the vacuum pump was closed for approximately 5 minutes to allowed the canister to stabilize before proceeding to the next level. This reduced the likelihood of ice build-up in the siphon line that could temporarily block the line and extend the time required to complete the vacuum drying. When the canister pressure reached 1.7 torr or less (Step 9.19), the 30 minute test was started (Step 9.20). The test was successful if the pressure remained below 2.8 torr for 30 minutes (Step 9.23), otherwise the vacuum drying process was re-initiated. The 2.8 torr limit was established to account for instrument error to ensure the 3.0 torr limit was met. Once the dryness criteria was met, the cask loading supervisor signed off on Step 9.23 that the Technical Specification 1.2.2 dryness criteria was met for the first dryness test. Section 10 of the procedure involved introducing helium to the canister to approximately 10 psig, after which the final vacuum drying was performed. Introducing helium, then performing a second vacuum drying, further ensured all water would be removed from the canister. Steps 10.21 thru 10.23 documented the completion of the final vacuum drying which required the pressure to remain below 2.8 torr for the 30 minutes. Completion of the test was signed-off by the cask loading supervisor and a quality control representative. The first canister was successfully dried to less than 2.8 torr at the end of 30 minute for both the initial drying and the final drying.

Two 0 - 50 pounds/square inch-absolute (psia) Ashcroft Model 2089 pressure gauges were used for the vacuum drying pressure test. Calibration records were reviewed for the two pressure gauges (IS Number 10820 and 10821). Both had been calibrated on June 28, 2010 at several pressure levels of 25, 15, 10, 5, 2.5 and 1.0 psia. Pressure Gauge 10820 required a +/- 0.10 psia tolerance. Pressure Gauge 10821 required a tolerance of +/- 0.40 psia. Both gauges successfully passed the required calibrations. Calibration records for the three pressure transducers available for use were reviewed. This included two MKS 750B pressure transducers (IS No. 10784 and 10786) and one MKS 722A pressure transducer (IS No. 10785). All three pressure transducers were calibrated on March 30, 2010. Acceptance criteria was +/- 5%. Calibrations were performed at 19, 5, 3 and 1 torr. All three pressure transducers successfully passed calibration.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (c) Nebraska Public Power District Calibration Table for

Ashcroft 2089 0-50 psia Gauge IS No. 10820, dated June 28, 2010, (d) Nebraska Public Power District Calibration Table for Ashcroft 2089 0-50 psia Gauge IS No. 10821, dated June 28, 2010, (e) Nebraska Public Power District Calibration Table for MKS 750B Pressure Transducer IS No. 10784, dated March 30, 2010, (f) Nebraska Public Power District Calibration Table for MKS 750B Pressure Transducer IS No. 10786, dated March 30, 2010, (g) Nebraska Public Power District Calibration Table for MKS 722A Pressure Transducer IS No. 10785, dated March 30, 2010

Category: Drying/Helium Backfill **Topic:** Vacuum Drying Time Limits

Reference: CoC 1004, Tech Spec 1.2.17 Amendment 9

Requirement: The time limit for vacuum drying a 61BT canister with a decay heat load of greater than 17.6 kW is 96 hours. For decay heat loads of 17.6 kW or less there is no time limit. If the canister cannot be vacuum dried to 3 mm Hg (torr) or less for 30 minutes or more within 72 hours, the canister must be backfilled with helium to 0.1 atmospheres or greater within the next 24 hours. The licensee must determine the cause of the failure to achieve vacuum drying pressure. After the cause is determined, the licensee is to initiate vacuum drying actions or unload the DSC within 30 days.

Observation: The first cask loaded at Cooper had a heat load of 11.3256 kW. This heat load did not require a time limit for vacuum drying because it was below 17.6 kW. Procedure 10.38, Attachment 10 had included steps for calculating the time limit restrictions for casks with heat loads greater than 17.6 kW. Vacuum drying time started with the completion of the canister pump down (Note above Step 1). Drying was required to be completed within 96 hours (Step 2.1). The 2.8 torr limit within 30 minutes must be demonstrated within 72 hours of pump-down completion (Note 1 above Step 2.2). If the vacuum drying pressure could not be achieved within 72 hours, then the canister must be filled with greater than or equal to 100 torr of helium within 24 hours (Note 2 above Step 2.2) and the cause of the problem determined, a condition report issued, and vacuum drying resumed (Steps 3.2 thru 3.8). The 100 torr value used in Procedure 10.38 equated to slightly over 0.1 atmosphere at standard temperature and pressure. Once the vacuum drying limit was met, Step 4 of the procedure documented the vacuum drying time. If the vacuum drying limit could not be met, Step 5.2.7.2 required the canister to be unloaded within 30 days.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4

Category: Emergency Planning **Topic:** Emergency Plan

Reference: 10 CFR 72.32(c) Published 2010

Requirement: For an ISFSI that is located on the site of a nuclear power plant licensed for operation, the emergency plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section.

Observation: The ISFSI was co-located with the Cooper nuclear power plant, and as such, was incorporated into the Part 50 emergency planning program. The ISFSI was mentioned in

the introduction to the emergency plan and one new emergency action level was added to Table 4.1-1 "Notification of an Unusual Event Emergency Action Level." Emergency Action Level EU1.1 was defined as damage to a loaded cask confinement boundary. This emergency action level was consistent with the guidance in the NRC endorsed document NEI 99-01 from the Nuclear Energy Institute. The Cooper emergency action levels were provided in the Emergency Plan Implementing Procedure (EPIP) 5.7.1. The procedure described EU1.1 as "An unusual event in this emergency action level is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage. Minor surface damage that did not affect the storage cask boundary was excluded." Other emergency action levels would be applicable to the ISFSI including security threats, radiological releases, and fires which could escalate the classification of the event into one of the other emergency classification levels of alert, site area, and general emergency.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) "Cooper Nuclear Station Emergency Plan," Revision 59, (c) Emergency Plan Implementing Procedure (EPIP) 5.7.1 "Emergency Classification," Revision 42, (d) Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 5

Category: Emergency Planning **Topic:** Emergency Plan Changes

Reference: 10 CFR 72.44(f) Published 2010

Requirement: Within six months of any change made to the emergency plan, the licensee shall submit a report containing a description of the changes to the appropriate regional office and to headquarters.

Observation: The licensee had incorporated into Procedure 0.29.1 the requirement to notify the NRC of changes to the site emergency plan. Procedure 0.29.1 required the Licensing Manager to notify the Director of the Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, US NRC, with a copy going to the NRC Region IV Office, of a description of any changes made to the Emergency Plan within 6 months after the change was made effective. An example of this requirement being implemented was provided in a letter from the Cooper Nuclear Station Licensing Manager to the US NRC dated September 9, 2010, providing a copy of Revision 59 of the Emergency Plan that was approved on September 7, 2010.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) "Cooper Nuclear Station Emergency Plan," Revision 59, (c) Administrative Procedure 0.29.1 "License Basis Document Changes," Revision 28, (d) Letter (NLS2010084) from David W. Van Der Kamp, Nebraska Public Power District to NRC Document Control Desk entitled "Cooper Nuclear Station Emergency Plan," dated September 9, 2010 [not publically available]

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Category: <u>Emergency Planning</u> **Topic:** <u>Emergency Training/Drills</u>

Reference: 10 CFR 50 App. E, Sect. F.1 Published 2010

Requirement: The emergency program shall provide for the training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees are familiar with their specific response duties.

Observation: The licensee had incorporated potential hazards associated with the ISFSI, and response actions for responding to an emergency at the ISFSI, into the training for emergency response personnel. The licensee had provided ISFSI system overview training to site emergency response personnel. The training covered the process, systems, components, and regulatory requirements associated with placing spent fuel in the ISFSI and included a description of the emergency action level associated with the ISFSI. On September 8, 2010, the licensee conducted a drill that included an event during the movement of a loaded canister to the ISFSI. The scenario involved the transport trailer breaking down at the sally port during a loss of offsite power. The transport trailer was repaired and moved to the ISFSI pad after which an onsite explosion occurred affecting a reactor safety system.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Lesson Plan No. OTH015-09-06 "ISFSI System Overview," Revision 00, (c) Attendance Sheet Report for Lesson OTH015-09-06, (d) Presentation Materials for Lesson OTH015-09-06, (e) Scenario Overview, Team 2a Off Hours Exercise, dated September 8, 2010

Category: <u>Emergency Planning</u> **Topic:** <u>Offsite Emergency Coordination</u>

Reference: 10 CFR 72.32(a)(15) Published 2010

Requirement: The applicant's emergency plans shall include a brief description of the arrangements made for requesting and effectively using offsite assistance onsite and provisions that exist for using other organizations capable of augmenting the planned onsite response.

Observation: The Cooper Nuclear Station Emergency Plan included a description of the arrangements for requesting and using offsite assistance for an onsite event involving a medical, radiological, and fire emergency at the Cooper site. These arrangements were applicable to the ISFSI operations. Section 7.9 "Medical Facilities and First Aid" discussed the arrangements for offsite emergency medical support including cases involving contamination. Section 7.6 "Fire Protection" and Section 6.4 "Corrective Action" discussed the onsite fire capability at the Cooper Nuclear Station and the available offsite fire support to augment the onsite fire brigade and fire fighting equipment. Section 5.4 "Participating Federal, State, and Local Agencies" discussed the offsite support for radiological assessment and protective measures decision making by the states of Nebraska, Missouri, Kansas and Iowa. Provisions for support during security events was discussed in the Cooper Nuclear Station Safeguards Plan. Appendix D "Letters of Agreement" of the emergency plan listed 25 organizations that had signed letters of agreement with the Cooper Nuclear Station to provide support during an emergency. These included security, fire response, medical, and radiological support. Selected letters of agreement for the local fire support and the local medical center were reviewed and found current. Each contained provisions for annual training visits to the Cooper site and participation in drills. Section 8.1.3 "Training for Participating Agencies" of the emergency plan stated that annual training to offsite agencies would be provided. As of

the time of this inspection, Cooper Nuclear Station had not provided training or site tours related to the ISFSI operations to offsite state and local emergency organizations. Several state and local emergency organizations were made aware that the ISFSI existed in the quarterly offsite agency emergency planning coordination meeting held March 20, 2009. An e-mail, dated September 16, 2010, was sent to the agencies on the emergency planning offsite agencies list by the Cooper Nuclear Station Emergency Preparedness Manager which provided the current schedule for the movement of casks to the storage pad and offered the agencies the opportunity to tour the ISFSI facility and receive an update on the status of the ISFSI at the next quarterly coordination meeting to be held in December 2010.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "Emergency Plan," Revision 59, (c) Emergency Planning Offsite Agency Coordination Meeting Agenda and Attendance Sheet, dated March 20, 2009, (d) Agreement No. 95A-C41 with the City of Auburn Volunteer Fire Department entitled "Fire Protection Support for Cooper Nuclear Station," dated December 5, 2005, (e) Agreement No. 4200001391 with the Community Medical Center, Inc., Falls City, Nebraska, entitled "Agreement to Provide Emergency Medical Transport Services for Cooper Nuclear Station's Radiological Emergency Response Plan," dated May 18, 2009, (f) E-mail from David N. Montgomery, Cooper Nuclear Station Emergency Preparedness Manager to the Contacts on the CNS Emergency Plan Offsite Agencies List entitled "Cooper Update for Dry Cask Storage," dated September 16, 2010

Category: Fire Protection **Topic:** Fire Hazards Analysis

Reference: CoC 1004, Tech Spec 1.1.1.5 Amendment 9

Requirement: The potential for fires and explosions, affecting the ISFSI and the loaded canister, must be addressed based on site-specific considerations.

Observation: The Cooper fire hazards analysis reviewed the fire and explosion hazards associated with the ISFSI and the haul path. Events that were evaluated included material handling equipment fires, spilled motor fuel pool fires, small vehicle fires adjacent to the horizontal storage modules, other materials that were planned to be stored within the ISFSI, and combustible materials along the transport route. The fire hazards analysis evaluated the hazards against the design criteria provided in the NUHOMS Updated Final Safety Analysis Report (UFSAR). For the transfer cask loaded with a canister, UFSAR Section 3.3.6 "Fire and Explosion Protection," Section K.4.6 "Thermal Evaluation for Accident Conditions," and Section K.11.2.10 "Fire and Explosion" provided information on the effects of a fire on the canister and the spent fuel. Section 3.3.6 stated that the horizontal storage modules and the canisters contain no flammable or explosive material. No fire suppression system was required within the boundaries of the ISFSI. Any response to a fire emergency would be provided by the plant fire brigade using portable fire suppression equipment. Specific analysis was provided in Appendix K "NUHOMS 61BT System" of the UFSAR for the canister design used at Cooper. Section K.1.2.2.1 "General Features" stated that the design temperature limit for the fuel cladding temperature for normal operations was 649 degrees F (343 degrees C). For short term accident conditions, short term off-normal conditions, and fuel transfer operations, the maximum fuel cladding design limit temperature was 1,058 degrees F (570 degrees C). These values were also provided in Table K.4-1 "NUHOMS-61BT

DSC Component Temperatures During Storage." Section K.4.6 "Thermal Evaluation for Accident Conditions" evaluated the affect on the canister from a hypothetical fire during transport of the canister. Section K.4.6.5 "Hypothetical Fire Accident Evaluation" analyzed a fire involving 300 gallons of diesel fuel during transport of a cask to the ISFSI. This scenario would bound fire scenarios associated with the horizontal storage module due to the large mass of the horizontal storage module and the vent configuration which provided protection for the canister. The cask contained spent fuel with the maximum decay heat load of 18.3 kW. Ambient temperature was assumed to be 125 degrees F with the solar shield in place on the transfer cask. The 15 minute fire completely engulfed the canister at a temperature of 1,475 degrees F. The resulting temperature on the canister surface was 499 degrees F. No direct comparison was provided in the calculation to the resulting temperature of the spent fuel cladding. However, the analysis for a blocked vent on the horizontal storage model provided a relationship between the canister shell temperature and the spent fuel cladding temperature. The blocked vent scenario was discussed in the UFSAR, Section K.4.1 "Thermal Evaluation Discussion," Section K.4.6.1 "Blocked Vent Accident Evaluation," Section 4.6.5 "Hypothetical Fire Accident Evaluation," and Table K.4-1 "NUHOMS 61BT DSC Component Temperatures During Storage." For the blocked vent, the temperature of the canister shell was calculated to reach 662 degrees F. This would result in the fuel cladding reaching a temperature of 809 degrees F, which was below the accident limit of 1,058 degrees F. As such, it can be seen that the 499 degrees F canister surface temperature for the fire would result in even a lower cladding temperature than the blocked vent scenario and well below the accident limit.

The ISFSI and the transport route from the reactor building to the ISFSI pad were within the plant's protected area. The Fire Hazards Analysis calculated the minimum distance between various hazards and the canister. These distances were incorporated into a series of graphs including Figure 4-1 "Minimum Required Distance of HSM from Structure Fires" which listed the acceptable minimum distance versus the square foot of the fire area for a building; Figure 4-2 "Minimum Required Distance of HSM from Large Structure Fires" which listed the minimum distance versus square foot of the fire area for large buildings up to 100,000 square feet; Figure 4-3 "Minimum Required Distance from Center of Hydrocarbon Pool Fires of Less than 25 Gallons;" Figure 4-4 "Minimum Required Distance from Center of Hydrocarbon Pool Fires Greater than 25 Gallons;" and Figure 4-5 "Minimum Required Distance from Center of Hydrocarbon Pool Fires of Much Greater than 25 Gallons." The basis and calculations for the figures was provided in Appendix B "Minimum Required Distances from Structure Fires" and Appendix C "Minimum Required Distances from Hydrocarbon Pool Fires." For the buildings, no credit was taken for fire suppression or detection systems when determining the minimum safe distances. Based on the values provided in the figures, Table 5.1 "Dimensions of Structures and Hazards with Line of Sight of Haul Path and HSM" was developed to evaluate the various structures and hazards located around the haul path and ISFSI at Cooper. The list included 27 structures and fire hazards that had been identified including various nearby buildings, the transformer yard, a 300 gallon gasoline tanks, the nearby diesel generator building, and hazardous materials storage cabinets. Of these, the craft change building and the technical support building did not meet the minimum distance requirements. The nearest structure to the ISFSI was the craft change building, which was 50 feet away. Based on Figure 4-1, the minimum distance for this 1,200

square foot building was 110 feet. For the first loading campaign of eight casks, the horizontal storage modules will be 175 feet away. As more casks are placed in the ISFSI, the horizontal storage modules will eventually get closer than the 110 foot limit. At that time, the licensee stated that a fire barrier would be constructed between the casks and the craft change building. The technical support building was a 3,000 square foot area building located 130 feet from the ISFSI and did not meet the distance requirements. Based on Figure 4-1, the minimum distance for this size building was 175 feet. The licensee determined that the distance was acceptable because the building had a fire detection system and suppression equipment that would bring the fire under control during the initial phases of the fire. In addition, the site fire brigade could readily respond to the fire.

For flammable liquids, the Fire Hazards Analysis listed a 300 gallon gasoline storage tank located 375 feet from the ISFSI. The minimum required distance based on the Fire Hazards Analysis calculations was 46 feet. For vehicles and bulk delivery trucks, Section 7.1.2 “Vehicle and Miscellaneous Equipment” evaluated the various fire hazards. Vehicles would be administratively kept at least 30 feet from the ISFSI. At 30 feet, Figure 4-4 provided a minimum volume for gasoline of 125 gallons and 175 gallons for diesel. For bulk delivery vehicles, the vehicles will be continuously manned and administratively controlled to stay away from the ISFSI. During transport of a cask to the ISFSI pad, a fire involving the diesel powered tractor pulling the cask on the trailer could occur. The tractor fuel tank was sized to a maximum of 300 gallons, which was the limit analyzed in Section K.4.6.5 of the UFSAR for the hypothetical fire. Wildfires were determined to not present a hazard to the ISFSI or haul path. The nearest vegetation to the ISFSI was 85 feet away.

Explosive hazards were evaluated in the Fire Hazards Analysis in Section 7.2 “Potential Explosion Hazards.” These included acetylene bottles, car gasoline tanks, lead acid batteries, portable propane tanks, and hydrogen storage tanks. One of the largest explosion hazards was five hydrogen storage tanks containing a total of 36,665 cubic feet of hydrogen. However, the storage tanks were located over 500 feet from the haul path and over 900 feet from the ISFSI with a number of buildings between the tanks and the haul path/ISFSI. All of the explosive hazards identified in the fire hazards analysis were evaluated and compared to the postulated external pressures that would result from tornado wind effects and tornado-generated missiles. All were enveloped by the design basis tornado.

Documents Reviewed:

(a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (c) ISFSI Fire Hazards Analysis, Revision 0, (d) American Concrete Institutes (ACI) 349 Code “Code Requirements for Nuclear Safety Related Concrete Structures,” Revision 1985

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Category: Fuel Selection/Verification **Topic:** Classifying Intact vs Damaged Fuel

Reference: Interim Staff Guidance (ISG) - 1 Revision 2

Requirement: An intact fuel assembly is a fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks which can be handled by normal means.

Observation: Damaged fuel was classified in Procedure 10.26 consistent with the definition in the NRC Interim Staff Guidance ISG-1, the NUHOMS Updated Final Safety Analysis Report (UFSAR), Table K.2-2 "Damaged BWR Fuel Assemblies Characteristics," and Technical Specification Table 1-1j "BWR Fuel Specification of Damaged Fuel to be Stored in the Standardized NUHOMS 61BT DSC." Procedure 10.26, Attachment 5 "Information Sheet," Step 1.3.2 defined damaged fuel as fuel assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks, assemblies missing fuel rods which are not replaced with dummy fuel rods, or those which cannot be handled by normal means. Seven classifications of fuel were defined in Procedure 10.26, Attachment 5, Section 1.3 "Definitions." Category 1 had four subcategories. Standard fuel assemblies with no evidence or suspicion of cladding penetration were classified as 1A. Dummy rods inserted in place of removed fuel rods were considered acceptable in this category as long as no fuel rod locations were left unfilled. Category 1B included fuel assemblies with structural damage but no evidence or suspicion of clad penetration. The structural damage would be limited such that the assembly was able to be handled by normal means. Category 1C included fuel assemblies with known or suspected cladding penetrations less than or equal to pinhole leaks or hairline cracks. These assemblies could have limited structural damage but must be able to be handled by normal means. Category 1D was fuel assemblies that may be classified as Category 1A, 1B, or 1C but with evidence or suspicion of clad or structural material degradation such that their ability to withstand normal and design basis events in storage, or the normal and hypothetical accident conditions of transport as intact fuel, was questionable. These fuel assemblies required additional evaluation prior to storage. Category 2 spent fuel was fuel assemblies without sufficient information in available records to provide a justifiable classification, or fuel assemblies with suspected cladding penetrations based on cycle information. Category 2 also contained fuel assemblies with suspected leaking fuel rods that were not examined, such that the number and size of the defects were unknown. Category 2 fuel assemblies required positive confirmation by nondestructive testing before they could be re-categorized into Category 1. Category 3 had two subcategories. Category 3A was spent fuel assemblies with known or suspected cladding penetrations larger than pinhole leaks or hairline cracks, but small enough to contain the gross fuel material. Category 3A also included assemblies with damage that precluded handling with normal means or had missing fuel rods which were not replaced with dummy fuel rods. Category 3B were fuel assemblies with known or suspected cladding penetrations which could allow the escape of significant quantities of fuel material. These fuel assemblies may have structural damage and cannot be handled by normal means due to clad conditions. The assemblies may also have missing rods.

For the first cask loading campaign, only spent fuel assemblies with no evidence or suspicion of cladding penetration or with cladding penetrations less than or equal to pinhole leaks or hairline cracks were permitted to be loaded. Procedure 10.36, which was used to develop the cask loading plan, stated in Step 2.3 that damaged fuel assembles were not to be loaded and only intact spent fuel identified as Category 1A and

1B, per Procedure 10.26, were allowed. Cooper had completed a review and classification of 1,940 spent fuel assemblies divided into 43 groups. Of these, 664 were classified as 1A, zero as 1B, zero as 1C, and 532 as 1D. There were 743 classified as Category 2 and one classified as Category 3B. No assemblies were classified as 3A on the list, however, the licensee informed the NRC inspectors that there were a total of four spent fuel assemblies that met the criteria to be classified as 3A or 3B. The one assembly (YJJ245) classified as 3B on the list was in Group 35 and had been in operating cycles 18, 19, and 20. Xenon offgas activity at the beginning of cycle 20 indicated that a fuel rod had failed during cycle 19. Fuel sipping of assembly YJJ245 at the end of cycle 20 identified the failed fuel rod as B3. This was confirmed by visual inspection which found several areas of secondary hydride damage near the fuel rod ends. Additional evaluation determined that the primary failure site was near the middle of the rod. Of the 532 spent fuel assemblies in the 14 groups that were classified as 1D, all but Group 39 had been classified as 1D because of increased corrosion and oxide spalling noted in fuel operating cycle 20 and beyond. No actual leakers had been identified in these groups using sipping. Group 39 consisted of one fuel assembly (YJJ246). The fuel assembly was classified as 1D because the bail handle was bent during insertion into the east fuel prep machine. The 743 assemblies classified as Category 2 were in 14 groups. Of these, increased offgas xenon values indicated fuel rod leaks in cycles 10, 11, 12, 17A, 18 and 19. Not all of these cycles produced clear evidence of xenon leaks during the particular cycle, but had elevated steady state levels of xenon similar to previous cycles which were higher than would be expected from a failure free cycle. During cycle 18, spikes from several iodine isotopes were noted during a series of power reductions. The iodine spikes were not observed during cycle 19. A small sampling of selected fuel assemblies from the cycles with elevated xenon levels were sipped or visually examined. No damaged fuel rods were found. All of these fuel assemblies will need further evaluation or sipping prior to fuel loading. Cycles 20, 21, and 22 were unique in that 546 fuel assemblies were exposed to noble metal injection but not hydrogen injection. This was unique to the industry and will require examination by the fuel vendor to determine the material condition of the clad and it's acceptability for dry cask storage in the NUHOMS casks.

Documents Reviewed: (a) NRC Interim Staff Guidance ISG-1 "Classifying the Condition of Spent Fuel for Interim Storage and Transportation Based on Function," Revision 2, (b) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (d) Nuclear Performance Procedure 10.26 "Fuel Classification of CNS Spent Fuel for Dry Storage and DOE Disposition," Revision 0, (e) Nuclear Performance Procedure 10.36 "Fuel Bundle Selection Process for Loading NUHOMS 61BT Dry Shielded Canister," Revision 3

Category: Fuel Selection/Verification **Topic:** Damaged Fuel Authorized for the 61BT Canister

Reference: CoC 1004 Tech Spec 1.2.1; Table 1-1j & 1-1d & 1-2q Amendment 9

Requirement: The standardized NUHOMS 61BT "Type C" canister is authorized to store damaged 7 X 7 and 8 X 8 General Electric or Exxon/ANF BWR fuel assemblies. The damaged fuel

assemblies shall be stored in the 2 X 2 compartments, and shall have the top and bottom caps installed. Damaged fuel may be stored with or without channels and must meet the parameters of Tables 1-1j, 1-1d and 1-2q.

Observation: No provisions had been made in the procedures for loading damaged fuel. Procedure 10.36, Step 2.3 restricted the loading of canisters to intact fuel classified as 1A or 1B. There were 664 spent fuel assemblies that had been classified as 1A and zero classified as 1B of the 1,940 fuel assemblies that had been classified. For the eight canisters planned for loading in the first loading campaign, a total of 8 x 61 = 488 assemblies were needed.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.26 "Fuel Classification of CNS Spent Fuel for Dry Storage and DOE Disposition," Revision 0, (c) Nuclear Performance Procedure 10.36 "Fuel Bundle Selection Process for Loading NUHOMS 61BT Dry Shielded Canister," Revision 2

Category: Fuel Selection/Verification **Topic:** Fuel Verification Prior to Loading

Reference: CoC 1004, TS 1.2.1; UFSAR 1004, Sect K.8.1.2.6 Amendment 9/Rev. 10

Requirement: Prior to insertion of a spent fuel assembly into the DSC, the identity of the assembly is to be verified by two individuals using an underwater video camera or other means. Read and record the fuel assembly identification number from the fuel assembly and check this identification number against the DSC loading plan which indicates which fuel assemblies are acceptable for dry storage

Observation: Procedure 10.36.1, Section 4.1 "Loading a DSC" provided the instructions for loading and performing the independent verification of the spent fuel. Step 4.1.3 required the fuel mover and the refuel floor supervisor to independently verify the identification number of each spent fuel assembly prior to removing it from the storage rack. The fuel mover and refuel floor supervisor were both on the refueling bridge during the independent verification, however, Step 4.1.3.1 stated that the fuel mover and refueling floor supervisor were not to discuss or collaborate on the actions to be performed. Attachment 2 "Fuel Assembly Identification Number Verification," of Procedure 10.36.1 was used by the fuel mover to locate the correct fuel assembly and move the refueling bridge to the correct location to retrieve the assembly. The refuel floor supervisor then moved the camera over the assembly that had been selected and read the fuel assembly number to the fuel mover. The refueling floor supervisor then verified the correct identification number using Attachment 2. Both the fuel mover and the refueling floor supervisor signed-off on Attachment 2. The independent verification process for the first canister loaded was observed by the NRC inspectors.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (c) Nuclear Performance Procedure 10.36.1 "Fuel Loading/Unloading of a Dry Shielded Canister," Revision 3

Category: <u>Fuel Selection/Verification</u> **Topic:** <u>Intact Fuel Authorized for the 61BT Canister</u>

Reference: CoC 1004 Tech Spec 1.2.1; Table 1-1c & 1-1d & 1-2q Amendment 9

Requirement: The standardized NUHOMS 61BT system is authorized to store intact, channeled or unchanneled, 7 x 7, 8 x 8, 9 x 9, and 10 x 10 General Electric or equivalent reload BWR fuel assemblies. The spent fuel assemblies shall meet the parameters of Tables 1-1c, 1-1d and 1-2q.

Observation: The fuel selected for the first canisters met the fuel selection criteria of Technical Specification 1.2.1 and associated tables. Procedure 10.36, Attachment 2 "BWR Fuel Specifications for Fuel to be Stored in the Standardized NUHOMS 61BT DSC" provided a list of the requirements from Technical Specification 1.2.1 and the associated tables. A completed Attachment 2 was generated for each fuel assembly which documented the characteristics of the fuel assembly against the requirements. For the first loading campaign of eight casks, only intact GE 8 x 8 spent fuel was planned for loading. The characteristics for each spent fuel assembly planned for loading in the first canister (CNS61B-007-A), as listed in Attachment 2, were reviewed by the NRC inspector and found to meet the technical specification requirements for storage in the 61BT canister. The first canister loaded was a Type A basket. Type A, B, and C baskets were available for the NUHOMS cask and were classified as A, B, or C based on the Boron-10 poison level in the canister basket. Technical Specification Table 1-1c "BWR Fuel Specification for the Fuel to be Stored in the Standardized NUHOMS 61BT DSC" listed the maximum allowable lattice average initial enrichment for a Type A basket as 3.7 wt % U-235. For the first canister, the highest maximum lattice average enrichment for the 61 spent fuel assemblies was 3.390 %. When comparing the assembly enrichment value to the requirement, a 0.04% uncertainty was added to the assembly enrichment value. Minimum cooling time was a function of initial enrichment and burnup. Technical Specification Table 1-2q "BWR Fuel Qualification Table for NUHOMS 61BT DSC" provided a table to determine the required cooling time. This table was duplicated in Procedure 10.36 as Attachment 3 "BWR Fuel Minimum Allowable Cooling Time for NUHOMS 61BT DSC." Acceptable minimum cooling times ranged from 4 years to 16 years. The licensee had evaluated each spent fuel assembly against the table and verified that the assembly met the minimum cooling time. For the first canister, cooling times for the spent fuel placed in the canister ranged from 15 years to 20.6 years. The maximum burnup allowed was 40 Gigawatt-Days/Metric Ton Uranium (GWD/MTU) based on Table 1-1q and the initial enrichment and cooling time of the spent fuel assembly. For the first canister, the highest burnup was for assembly LYU391 with 37.505 GWD/MTU. This assembly had a cooling time of 15 years and an initial enrichment of 3.39 wt % U-235. To account for uncertainty, 1.05 GWD/MTU was added to the calculated burnup value for the assembly when comparing the burnup value to the technical specification limit. Individual assemblies were limited to 300 watts per Technical Specification Table 1-1c for maximum decay heat. This individual limit times 61 assemblies allowed in a canister resulted in a total limit to the canister of 61 x 300 = 18.3 kW. Of the 61 assemblies in the first canister, the highest decay heat was 234.9 watts for assembly LYU429. Technical Specification Table 1-1d "BWR Fuel Assembly Design Characteristics for the NUHOMS 61BT DSC" listed the acceptable types of fuel allowed in the 61BT canister. The fuel assemblies for the first cask loading were a mixture of GE-Barrier 8 x 8 (GE7B) assemblies, GE9 assemblies, which were also 8 x 8 assemblies, and GE-Pressurized 8 x 8 assemblies (GE-Pres). Table 1-1d listed these

assembly designs as acceptable.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.36 "Fuel Bundle Selection Process for Loading NUHOMS 61BT Dry Shielded Canister," Revision 2, (c) Nuclear Performance Procedure 10.48 "Caseworks Input Data Generation," Revision 1

Category: Fuel Selection/Verification **Topic:** Material Balance, Inventory, and Records

Reference: 10 CFR 72.72(a) Published 2010

Requirement: Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all SNM with quantities specified in 10 CFR 74.13(a)(1).

Observation: Records of special nuclear material (SNM) transfers and inventories were required by Procedure 10.21. The item controlled areas (ICA) were defined in Attachment 6 "Information Sheet," Step 2.2.11 as physical areas that may be designated by reactor engineering which are clearly separate from all other areas and are within the restricted area of the plant site. The boundaries of the item controlled areas are intended to provide control points for movement of SNM. Step 2.2.11 identified the dry shielded canister (DSC) and the horizontal storage modules as designated item controlled areas. Procedure 10.21 provided instructions for moving SNM between item controlled areas and provided Attachment 1 "SNM Transfer Form" to document the change from one item control area to another. For the spent fuel loaded in the canister, Attachment 1 showed the spent fuel rack location the fuel assembly had been stored in and the slot number in the canister where it was placed. Annually, an inventory of the SNM was required by Section 4 "Twelve Month SNM Inventory" of Procedure 10.21. Attachment 3 "SNM Physical Inventory" provided a form to document completion of the annual inventory. Step 11.7 "SNM Status Report" of Procedure 10.21 required completion of DOE/NRC Form 741/742 by May 31 of each year and submittal of the report to the NRC.

Cooper Nuclear Station had previously shipped spent nuclear fuel to the General Electric Morris Facility in Morris, Illinois for storage in their wet independent spent nuclear fuel storage installation. NUREG-0725, Table 3-1 "Number of Shipments and Quantity of Spent Fuel Shipped from 1979 to 2007" listed 30 shipments and a total of 194,546 kilograms of spent fuel. The fuel had been shipped to Morris between 1984 and 1989 according to Cooper's Engineering Evaluation 09-011, Section 4.1.1 "Background "using the GE IF-300 transportation canister. GE Transaction Reports (NRC Form 741) taken from the Cooper microfilm records indicated 31 shipments for a total of 1074 fuel assemblies. Of these, all but three shipments contained 36 fuel assemblies. One contained 30 assemblies and two contained 18 assemblies.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Nuclear Performance Procedure 10.21 "Special Nuclear Materials Control and Accountability Instructions," Revision 41, (c) NUREG-0725 "Public Information Circular for Shipment of Irradiated Reactor Fuel," Revision 15, (d) Administrative Procedure 0.8, Attachment 5 "72.48 Screening Form" for Activity Engineering Evaluation EE-09-011 "Review of 72.212 Report and Haul Path Hazards Analysis," dated October 5, 2010

Category: General License **Topic:** Changes, Tests, and Experiments

Reference: 10 CFR 72.48(c)(1) Published 2010

Requirement: A licensee can make changes to their facility or storage cask design if certain criteria are met as listed in 10 CFR 72.48.

Observation: The licensee had established a process in Procedure 0.8 to make changes to their dry cask storage program in accordance with 10 CFR 72.48. This procedure was also used for changes in accordance with 10 CFR 50.59. The 72.48 and 50.59 process was used to make changes to structures and programs being implemented at the Cooper site in accordance with the provisions of the Part 50 reactor license and the provisions of a general ISFSI license in Part 72. Step 4.5.3.1 of Procedure 0.8 stated that changes to the cask safety analysis report can only be made by the certificate holder (Transnuclear). As such, if a 72.48 evaluation indicated that a change was required to the NUHOMS license, technical specifications, or the NUHOMS Updated Final Safety Analysis Report (UFSAR), then a request had to be sent to Transnuclear requesting the change. The process to evaluate an issue was started by completing Attachment 2 "Applicability Determination Form." This form had several questions that directed the user to the correct process for the proposed activity and helped determine if a 72.48 or 50.59 screening or evaluation was appropriate. If a screening was required, Attachment 3 "50.59 Screen Form" and/or Attachment 5 "72.48 Screen Form" would be completed. Based on a series of questions in these two attachments, the user would determine if an evaluation was required. The evaluations were performed using Attachment 4 "50.59 Evaluation Form" and/or Attachment 6 "72.48 Evaluation Form." A conclusion was reached after the evaluation as to whether the activity could be performed in accordance with plant procedures or required a license amendment prior to implementation. For Attachment 6 related to the 72.48 evaluation, if the activity required a license amendment, a request was made to the certificate holder (Transnuclear) requesting the amendment. Guidance for completing the 50.59 and 72.48 forms was provided in Attachment 7 "50.59 Quality Criteria" and Attachment 8 "72.48 Quality Attachment." A number of 72.48 and 50.59 screenings were conducted. The screenings included such topics as the upgrade of the haul path road, ISFSI security design, reactor building crane upgrade, the 70-ton limit on the crane, ISFSI electrical design, the 72.212 report, and revisions to ISFSI related procedures. No 72.48 evaluations or license amendments were required as the result of the screenings. A 50.59 evaluation was performed related to the reactor building crane.

Selected 50.59 and 72.48 screenings were reviewed. The 72.48 screening No. EE 09-011 reviewed the design parameters and licensing activities associated with the NUHOMS 61BT canister and the Model 202 horizontal storage module and referenced that the 10 CFR 72.212 Evaluation Report for the Cooper Nuclear Station site found that the Cooper site was bounded by the design features for the NUHOMS system. The 72.48 screening determined that the activities reviewed in the screening did not require a 72.48 evaluation. The 50.59 screening for Activity CED 6023100 reviewed the reactor building crane upgrade to improve the reliability of the crane. The upgrade replaced a significant number of components including the main and auxiliary hoist motors, bridge and trolley controllers and motors, main and auxiliary hoist brakes, and the bridge and trolley primary brakes. In addition, enhancements were made to a number of

components including the load path limit switches, trolley bus bars, runway conductor bus-bar system, and cable power feed system. A new refuel floor radio operated control system was added and pull-points were added to the bridge and trolley. The screening resulted in a "yes" answer to two screening questions. Question 5.1 "Does the proposed activity involve a change to a structure, system, or component that adversely affects a Cooper Updated Safety Analysis Report (USAR) described design function," and Question 5.2 "Does the proposed activity involve a change to a procedure that adversely affects how USAR described structure, system, and component design functions are performed or controlled" received a "yes" answer. As such, a 50.59 evaluation was required. Evaluation No. 2007-0003 was performed. The evaluation determined that there were no accidents described in the USAR that were directly or indirectly impacted by the crane modifications. The crane was not discussed anywhere in the USAR as an initiator either in normal operations or any failure modes related to any accidents. The evaluation concluded that a license amendment was not required because the modification to the reactor building crane could not cause an accident, introduce the possibility of a change in the consequences of an accident, introduce new failure modes due to their failures, nor introduce any new accidents or scenarios not already bounded by the safety analysis and did not revise or replace a USAR described evaluation methodology that was used in establishing the design basis or used in the safety analysis.

The Cooper Nuclear Station crane had originally been rated at 100 tons. Because of the weight of the loaded canister, the crane rating needed to carry the loaded canister was 108 tons. Physical modifications were made to the crane to increase the rating to 108 tons. The modifications were reviewed in a 50.59 screening for Activity CED 6028740. The modifications included replacing the main hoist motor variable frequency drive, replacing the lower load cell connecting pins, increasing the weld size of two welds on the equalizer bar support plate, installing girder stiffening bars, increasing the size of the wire ropes, and several other modifications. New safety factors were calculated based on increasing the load from 100 tons to 108 tons. A review of each component of the crane was completed and a table developed which compared the new safety factor to the old safety factor and the licensing basis safety factor. The safety factors were from the Crane Manufacturers Association of America (CMAA) Guide #70. For non-redundant load bearing parts where full redundant features were not feasible, a minimum safety factor of 8.2 was required based on the ultimate strength of the material. Section X-4.4.1 "Single Failure Considerations" in the Cooper Nuclear Station USAR specified the 8.2 safety factor. For redundant trolley components and mechanical bridge components, the stress criteria was 5 to 1. The 50.59 screening of the crane up-rate determined that a 50.59 safety evaluation was not required. The crane was not included in the technical specifications for the Part 50 license and the physical modifications to up-rate the crane were made to ensure that the crane could continue to perform its design function at an increased rated load of 108 tons.

The ability of the reactor building superstructure to support the crane loading at 108 tons during normal and safe shutdown earthquakes was evaluated in the 50.59 screening for Activity EE 10-024. Engineering Evaluation (EE) 10-024 reviewed the original analysis for the building and performed a new 3D finite element model analysis to confirm the adequacy of the superstructure. The screening of the analysis determined that the 50.59 screening criteria had been met and that the current licensing basis calculations of record

bounded the re-rate from 100 tons to 108 tons. The newer and more complex methodologies (3D) confirmed the analysis of the original calculations.

Provisions had been included in the contract with Transnuclear for copies of 72.48 reviews completed by Transnuclear to be provided to Cooper. In addition, Transnuclear submitted a biennial report to the NRC listing all 72.48 evaluations completed. Cooper Nuclear Station reviewed the 72.48 evaluations performed by Transnuclear to verify that issues that could affect the Cooper dry cask storage system were adequately addressed. The Transnuclear biennial reports of the 72.48 evaluations completed between February 2006 to July 2008 and the period July 2008 to July 2010 were reviewed by the NRC inspectors. The 72.48 evaluations included several issues that related to the Cooper ISFSI including changes that had been incorporated into Revision 10 of the Transnuclear UFSAR, the introduction of the horizontal storage module (HSM) Model 202 for use with the various canisters including the 61BT used at Cooper, clarification of the transfer cask external contamination limits, the use of solid upper and lower trunnions as an alternative to the multi-piece trunnion design for the OS197 and OS197H transfer casks, further analysis of situations where the gap between the canister shell and the basket was less than the minimum specified in the design drawing due to local distortion of the shell, application of the design basis tornado and missile spectrum used for the horizontal storage module to the OS197 transfer cask, which previously had used lower values, further analysis of the location of the rails in the horizontal storage module in relation to the neutron absorber/insert centerlines, and the impact of NRC Information Notice 2009-23 related to thermal performance of the various Transnuclear canisters including the 61BT canister. No issues were identified that adversely affected the Cooper storage systems.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Administrative Procedure 0.8 "10CFR50.59 and 10CFR72.48 Reviews," Revision 18, (c) Administrative Procedure 0.8, Attachment 5 "72.48 Screening Form" for Engineering Evaluation EE-09-011 "Review of 72.212 Report and Haul Path Hazards Analysis," dated October 5, 2010, (d) Procedure 08, Attachment 3 "50.59 Screen Form," for Activity CED 6023100 "Reactor Building Crane Upgrade," dated September 20, 2007, (e) Administrative Procedure 0.8, Attachment 4 "50.59 Evaluation Form," for Activity 2007-0003 "Reactor Building Crane Upgrade," dated September 27, 2007, (f) Administrative Procedure 0.8, Attachment 3 "50.59 Screen Form," for Activity "Reactor Building Crane Re-Rate," dated July 7, 2010, (g) Administrative Procedure 0.8, Attachment 3 "50.59 Screen Form" for Activity EE 10-024 "Reactor Building Crane Re-Rate Evaluation," dated July 13, 2010, (h) Administrative Procedure 08 "10CFR50.59 and 72.48 Reviews," Attachment 3 "50.59 Screen Form," for Activity CED 6028740 "Re-Rate Reactor Building Crane from 100 Tons to 108 Tons," dated July 7, 2010, (i) Letter (E-26771) from D. Shaw, Transnuclear to NRC Document Control Desk entitled "Submittal of Biennial Report of 72.48 Evaluations Performed for the Standardized NUHOMS System, CoC 1004, for the Period 02/04/06 to 07/25/08, Docket 72-1004," dated July 25, 2008 [NRC ADAMS Accession No. ML082110243], (j) Letter from D. Shaw, Transnuclear to NRC Document Control Desk entitled "Submittal of Biennial Report of 72.48 Evaluations Performed for the Standardized NUHOMS System, CoC 1004, for the Period 07/26/08 to 07/23/10, Docket 72-1004," dated July 23, 2010 [NRC Adams Accession No. ML102080482], (k) Crane Manufacturers Association of America

(CMAA) Guide #70 "Top Running and Gantry Type Multiple Girder Electric Overhead Traveling Cranes," released 1971, (l) NRC Information Notice 2009-23 "Nuclear Fuel Thermal Conductivity Degradation," issued October 8, 2009 [NRC ADAMS Accession No. ML091550527], (m) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24, (n) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10

Category: General License **Topic:** Evaluation of Effluents/Direct Radiation

Reference: 10 CFR 72.212(b)(2)(i)(C) & 10 CFR 72.104(a) Published 2010

Requirement: The general licensee shall perform a written evaluation prior to use that establishes that the requirements of 10 CFR 72.104 "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI" have been met. 10 CFR 72.104 requires the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ during normal operations and anticipated occurrences,

Observation: The ISFSI was located within the protected area of the reactor site. Results of the licensee's evaluation of offsite doses were presented in the 10 CFR 72.212 Evaluation Report, Section 8 "10 CFR 72.212(b)(2)(i)(C) - Radioactive Materials in Effluents and Direct Radiation." The NUHOMS Updated Final Safety Analysis Report (UFSAR), Sections K.11.1.3 "Off-Normal Release of Radionuclides" and K.11.1.4 "Radiological Impact from Off-Normal Operations" stated that abnormal conditions were analyzed and found to not affect the shielding for the storage system, since none of the events would result in stresses on the sealed canisters in excess of allowable stresses. Thus, no breach of the canister would occur and the offsite dose contributions for the ISFSI from abnormal occurrences would not result in any increase over the doses from normal operations.

The distance from the ISFSI to the nearest public access was approximately 800 meters at a boundary location north of the ISFSI. Results of the licensee calculations showed that for a fully loaded ISFSI containing 52 loaded canisters in a 2 X 26 array, the annual direct radiation contribution due to the ISFSI operations would be 0.07 mrem with dose due to ISFSI effluents of 0.00 mrem. 10 CFR 72.104 required that radioactive material in effluents and direct radiation from the ISFSI and reactor facility be combined for comparison with regulatory limits. The results of combining the reactor plant operations with ISFSI operation to obtain the maximum annual offsite dose contribution were presented in the 10 CFR 72.212 Evaluation Report in Table 8.2-4 "Worse Case Annual Offsite Dose Contribution for ISFSI and Cooper Nuclear Station Plant." From the table, the whole body dose was determined to be 1.13 mrem, well below the 10 CFR 72.104 limit of 25 mrem per year. The thyroid dose from reactor operations was 0.14 mrem compared to the regulatory limit of 75 mrem. The dose for other critical organs from reactor operations was 1.47 mrem compared to the regulatory limit of 25 mrem per year. Thus, exposure to any real individual beyond the controlled area boundary was well below the limits of 10 CFR 72.104(a).

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Updated Final

Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10

Category: General License **Topic:** Flood Conditions

Reference: CoC 1004, Tech Spec 1.1.1.4 Amendment 9

Requirement: The site specific analyzed flood condition shall be no greater than 15 feet per second water velocity and a height of 50 feet of water (full submergence of the loaded HSM). This evaluation may be included in the 72.212(b) evaluation report.

Observation: The analyzed flood conditions for the Cooper Nuclear Station ISFSI site were bounded by the 15 feet per second (ft/sec) water velocity and a flood height of 50 feet of water specified in Technical Specification 1.1.1.4. A summary of the flooding potential for the site was provided in the 10 CFR 72.212 Evaluation Report, Section 10.2.2 "Flood." The river flow data from the Cooper Updated Safety Analysis Report (USAR), Table II-4-2 "Analytical Determination of River-Stage Discharge," can be used to estimate river flow rate at various postulated flood elevations. A flood at an elevation of 904 feet above mean sea level (MSL) resulted in a channel flow rate of approximately 13.6 ft/sec. However, river velocity over the total area, which was the river channel area plus the overbank section area where the ISFSI was located, resulted in a calculated flow rate of 2.4 ft/sec. Therefore, the flow velocity of floodwaters at the edge of the ISFSI basemat would be less than the NUHOMS® design criteria of 15 ft/sec.

The USAR, Section II-4.2.2.1 "Probable Maximum Flood" identified that the probable maximum flood for the Cooper site would occur at 903 feet above MSL. The elevation of the ISFSI basemat was 903.6 feet above MSL. Therefore, the top of the basemat was above the elevation of the probable maximum flood for the site. In addition, during initial licensing for the reactor facility, the effect of wave action, due to a sustained wind of 45 mph, was analyzed and resulted in a wave height of 6.7 feet. When considering the potential for wind-generated waves during the probable maximum flood, the height of floodwaters would still be significantly below the 50 ft flood height above the ISFSI pad assumed in the analysis of the NUHOMS® system. During the severe flooding in June/July 2011, Cooper Nuclear Station issued a Notification of Unusual Event emergency classification on June 19, 2011, due to the elevation of the Missouri River reaching 899 feet MSL. By July 12, 2011, the water level had dropped to 895.8 feet MSL. During June and July, the water level did not reach the ISFSI pad at the 903.6 foot (MSL) level. The worst previous flood had occurred in July 1993 which reached a crest height of 900.8 feet MSL at Cooper. Extensive flooding had occurred prior to installing the current upstream river controls in 1952 which reached 899 feet MSL.

The licensee had evaluated dam failures in the USAR, Section II-4.2.2.1 and II-4.2.2.2 "Site Flooding Protection." There were six dams on the Missouri River with the Gavins Point Dam the closest at 275 miles upstream. The USAR stated that failure of the upstream dams was not considered credible by the Corp of Engineers. Section II-4.2.2.2 stated that failure of one or more of the upstream dams combined with the maximum natural flood could result in flood levels at the Cooper site of 905 - 906 feet MSL.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS

Accession No. ML071070570], (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24

Category: General License **Topic:** Initial Compliance Evaluation Against CoC

Reference: 10 CFR 72.212(b)(2)(i)(A) Published 2010

Requirement: A general licensee shall perform written evaluations, prior to use, that establish that the conditions set forth in the Certificate of Compliance have been met.

Observation: Cooper Nuclear Station performed a written evaluation to document that the conditions set forth in the Certificate of Compliance had been met. Appendix A "CNS Certification No. 1004 Amendment 9 Compliance Evaluation" of the 10 CFR 72.212 Evaluation Report listed all of the requirements of the Certificate of Compliance, Number 1004, Amendment 9, for the Standardized NUHOMS System. The requirements were presented in tabular form along with a description of how compliance was demonstrated at the Cooper Nuclear Station. Specific sections of the 10 CFR 72.212 Evaluation Report or specific procedures and reports were cross-referenced to demonstrate compliance.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0

Category: General License **Topic:** Initial Compliance Evaluation Against FSAR

Reference: 10 CFR 72.212(b)(3) Published 2010

Requirement: The general licensee shall review the FSAR referenced in the CoC and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analysis of earthquake intensity and tornado missiles, are enveloped by the cask design basis considered in these reports. The results of this review must be documented in the evaluation made in 10 CFR 72.212(b)(2).

Observation: The licensee determined, through review of the Cooper Updated Safety Analysis Report (USAR), that the reactor site design basis parameters were enveloped by the NUHOMS cask design basis parameters. This review was documented in the 72.212 Evaluation Report, Section 10.0 "10 CFR 72.212(b)(3) - Reactor Site Parameters." The site parameters and conditions that were identified that required evaluation to demonstrate that the conditions assumed in the NUHOMS Updated Final Safety Analysis Report (UFSAR) and the certificate of compliance were bounded by the Cooper Nuclear Station site conditions included weather conditions, flooding, seismic conditions, and the potential for fires and explosions.

Weather conditions included extreme temperatures, flooding, tornados and high winds, ice and snow, and lighting. The evaluation of extreme temperature conditions is provided in these Inspector Notes under the Category: General License and the two Topics: Site Average Temperature and Site Temperature Extremes. Extreme temperatures at the Cooper site were found to be bounded by the design basis weather extremes discussed in the NUHOMS UFSAR. Flooding is discussed in these Inspector Notes under the Category: General License and the Topic: Flood Conditions. Flooding

conditions at the Cooper site were found to be bounded by the flooding analysis provided in the UFSAR.

High winds, tornadoes, and tornado driven missiles were discussed in the 10 CFR 72.212 Evaluation Report in Section 10.2.3 "Extreme Winds, Tornado, and Tornado Missiles." The NUHOMS UFSAR used NRC Regulatory Guide 1.76 “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants” for the design basis tornado wind intensities. Regulatory Guide 1.76 described tornado wind intensity regions for the contiguous United States. Tornado intensity Region I, the region with the highest tornado intensity values in the country, were the values used as the design basis for the NUHOMS cask system. The design basis tornado driven missile impact used in the NUHOMS UFSAR was based on the criteria provided in NUREG-0800 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.” The NUHOMS UFSAR tornado was defined in Section 3.2.1 "Tornado and Wind Loading" and Section 8.2.2 "Tornado Winds/Tornado Missiles." The NUHOMS design basis tornado had a maximum wind speed of 360 mph, a maximum translational speed of 70 mph, and a rotational speed of 290 mph. Calculations in Section 8.2.2.2 "Accident Analysis" showed that the winds associated with the design basis tornado were not capable of sliding or turning over a horizontal storage module or turning over the transport trailer loaded with the transfer cask. The Cooper Nuclear Station USAR, Section II-3.2.2 "Wind" defined the site design basis tornado as 300 mph tangential wind velocity with a 60 mph transverse velocity. As such, the Cooper design basis tornado was enveloped by the NUHOMS tornado winds. For tornado driven missiles, the NUHOMS UFSAR discussed the missiles and the analysis of the impact on the horizontal storage module in several sections including Section 3.2.1 "Tornado and Wind Loadings," Section 8.2.2 "Tornado Winds/Tornado Missiles," Appendix P, Section P.11.2.3.2.1.2 "Massive Missile Impact Analysis," and Appendix V, Section V.11.2.3.2.1 "HSM Model 202 Missile Impact Analysis" which referenced the analysis performed in Section P.11.2.3.2.1. The 72.212 Evaluation Report, Table 10-1 "Comparison of HSM Model 202 and CNS Design Tornado Missile Spectra" listed the tornado missiles discussed in the various NUHOMS UFSAR sections listed above plus missiles evaluated for the HSM-H design. These missiles included a 1,500 lb wooden telephone pole 35 feet long traveling at 294 feet/sec (200 mph), an armor piercing artillery shell 8 inches in diameter weighting 276 lbs traveling at 185 ft/sec (126 mph), a 12 inch diameter steel pipe 30 feet long weighting 1,500 lbs traveling at 205 ft/sec (140 mph), and a 4,000 lb automobile traveling at 195 ft/sec (133 mph). The Cooper USAR, Section XII-2.3.3.2.2 "Tornado Generated Missiles" defined four missiles that were used as the design basis missiles for the Cooper site. These included the 35 foot telephone pole at 200 mph, a one ton (2,000 lb) missile such as a compact type automobile traveling at 100 mph, a 2 inch heavy pipe 12 feet long (no velocity given), and any other missiles resulting from failure of a structure or component. Since missile mass and velocity are two key elements in the ability of the missile to cause damage, the tornado driven missiles analyzed in the NUHOMS UFSAR bounded those defined in the Cooper USAR as site design basis missiles.

The effects of heavy snow and ice on the horizontal storage modules was evaluated in the NUHOMS UFSAR, Section 3.2.4 "Snow and Ice Load." The snow and ice load design basis for the horizontal storage modules was derived from the American National

Standards Institute (ANSI) A58.1-1982, "Minimum Design Loads for Buildings and Other Structures." The maximum assumed 100 year roof load, specified for most areas of the continental United States, for an unheated structure, was 110 lbs/sq ft. For conservatism, a total live load of 200 lbs/sq ft was used in the horizontal storage module analysis to envelope all postulated live loadings, including snow and ice. The Cooper USAR, Section II-3.1.3 "Precipitation" provided information related to snow fall for the region. The USAR stated that snowfall was about 25 inches in the average season. The largest recorded amount was 59.4 inches that fell during the 1914-15 season. Much of the snow is light and melts rapidly. However, at times a considerable amount accumulates on the ground. The greatest recorded snow depth was 21 inches in February 1965. Assuming the 21 inches was solid ice, the maximum live load would be 100.1 lbs/sq ft. This would be below the 200 lbs/sq ft used for the horizontal storage module analysis. Snow and ice loads for the transfer cask with a loaded canister were considered negligible due to the smooth curved surface of the cask, the heat given off by the spent fuel assemblies, and the infrequent short term use of the cask.

The potential for lightning damage to the spent fuel while stored at the ISFSI was discussed in the NUHOMS UFSAR, Section 8.2.6 "Lightning." Lightning was not considered a hazard to the ISFSI. Section 8.2.6.2 stated that the current discharge from a lightning strike would follow a low impedance path and would not create damage to the horizontal storage module from heat or mechanical forces. Appendix V "NUHOMS HSM Model 202", Section V.11.2.5 "Lightning" stated that lighting protection equipment may be installed on the horizontal storage module. Lightning protection was added to the horizontal storage modules at the Cooper ISFSI via conductors to ground plate attachments provided in the Model 202 design. Lightning protection was also provided to the security fence and security systems.

Analysis of the earthquake potential at the Cooper site and comparison against the design basis for the NUHOMS cask system was provided in the 72.212 Evaluation Report in Section 10.2.4 "Earthquake Intensity/Seismic Acceleration" and is discussed in these Inspector Notes under the Category: General License and the Topic: Seismic Acceleration. The evaluation determined that the Cooper ISFSI was bounded by the NUHOMS cask system design for earthquakes.

Fires and explosions were discussed in the 72.212 Evaluation Report in Section 10.2.6 "Fire and Explosion." A discussion of this topic is provided in these Inspector Notes under the Category: Fire Protection and the Topic: Fire Hazards Analysis. The evaluation found that the Cooper ISFSI was bounded by the fires analyzed in the NUHOMS UFSAR.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (d) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24

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Category: General License **Topic:** Initial Evaluation Against Part 50 License

Reference: 10 CFR 72.212(b)(4) Published 2010

Requirement: Prior to use of the general license, determine whether activities related to storage of spent fuel involve a change in the facility technical specifications or require a license amendment for the facility pursuant to Part 50.59(c)(2). Results of this determination must be documented in the evaluation made in 10 CFR 72.212(b)(2).

Observation: Cooper Nuclear Station performed the required evaluation of the activities related to the storage of spent fuel against their Part 50 license and technical specifications. The evaluation was documented in the 72.212 Evaluation Report, Section 11.0 "10CFR72.212(b)(4) - Facility License and Technical Specifications." The review determined that the dry cask storage activities could be conducted at the Cooper site using the NUHOMS cask system without changes to the Part 50 license or technical specifications. The primary review was conducted as part of the 50.59 and 72.48 screening for Activity EE 09-11. This activity reviewed the broad range of issues associated with implementing a Part 72 general license at the site and reviewed the cask specific parameters associated with the NUHOMS 61BT canister and the HSM-202 horizontal storage module.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Administrative Procedure 0.8, Attachment 5 "72.48 Screening Form" for Engineering Evaluation EE-09-011 "Review of 72.212 Report and Haul Path Hazards Analysis," dated October 5, 2010

Category: General License **Topic:** Lightning Damage

Reference: CoC 1004, Tech Spec 1.1.1.7 Amendment 9

Requirement: The potential for lightning damage to any electrical system associated with the standardized NUHOMS system should be addressed based on site specific considerations. This evaluation may be included in the 72.212(b) evaluation report.

Observation: The potential for lightning damage to electrical systems associated with the ISFSI was addressed by adding lightning protection for the horizontal storage modules and ancillary support systems, based on standard lightning protection codes. Section 10.2.5 "Lightning," of the 10 CFR 72.212 Evaluation Report and the NUHOMS Updated Final Safety Analysis Report (UFSAR), Section 8.2.6 "Lightning," discussed lightning protection for the horizontal storage modules by providing lightning protection on the horizontal storage modules or the structures surrounding the horizontal storage modules. Lightning protection surrounded the perimeter of the ISFSI and interconnected the perimeter structures and equipment, including the intrusion detection equipment, security systems, and security fencing. Air terminals and lightning arrestors were installed on the security lighting located around the perimeter of the ISFSI and on the camera towers provided for the ISFSI. Additional lightning protection was provided for the horizontal storage modules via conductors to ground plate attachments as provided for in the HSM Model 202 design. The NUHOMS® UFSAR, Appendix V "NUHOMS HSM Model 202", Section V.11.2.5 "Lightning" stated that lighting protection equipment may be installed on the horizontal storage module.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS

Accession No. ML071070570], (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10

Category: General License **Topic:** Program Review - RP, EP, QA, and Training

Reference: 10 CFR 72.212(b)(6) Published 2010

Requirement: The general licensee shall review the reactor emergency plan, quality assurance program, training program and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals.

Observation: The 72.212 Evaluation Report, Section 13 "10 CFR 72.212(b)(6) - Program Effectiveness" provided a summary of the licensee's review of the existing reactor emergency plan, quality assurance program, training program, and radiation protection program. Changes were made to the programs to incorporate the dry cask storage program. The emergency plan was revised to include a new Notification of Unusual Event emergency action level for damage to a loaded cask confinement boundary. This was the only change required to the emergency plan. The quality assurance program used for the Part 50 activities under 10 CFR Part 50, Appendix B was reviewed and revised to incorporate the dry cask storage program. The quality assurance program changes incorporated controls for important-to-safety items. Procedures 0.19 was revised and Procedure 0.19.1 issued to address the process for classifying important-to-safety items.

The Part 50 training program was used to provide training for the dry cask storage program. Several new training modules were developed specific to cask loading activities and a training qualification program developed specific to work activities associated with dry cask storage. The training program is discussed in these Inspector Notes under the Category: Training. The radiation protection program was reviewed to determine if changes were needed for the ISFSI activities. The radiation protection program being implemented for the Part 50 program was determined to be applicable to the activities planned for dry cask storage. Special training was provided to the radiation protection staff specific to the potential radiation issues that would be associated with loading a canister and storing the spent fuel on the ISFSI pad.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0 (draft), (c) Letter (NLS2007076) from David W. Van Der Kamp, Nebraska Public Power District to the Director, NRC Spent Fuel Project Office entitled "Notification of Intent to Apply Previously Approved 10 CFR 50 Quality Assurance Program to Independent Spent Fuel Storage Installation Activities Cooper Nuclear Station, Docket No. 50-298, DPR-46," dated November 1, 2007 [NRC ADAMS Accession No. ML073120011], (d) Cooper Nuclear Station "Emergency Plan," Revision 59, (e) Administrative Procedure 0.19 "Equipment and Record Functional Location File Program," Revision 25, (f) Administrative Procedure 0.19.1 "Quality Assurance Program Applicability to Dry Fuel Storage," Revision 1

Category: General License **Topic:** Revisions to 72.212 Analysis

Reference: 10 CFR 72.212(b)(2)(ii) Published 2010

Requirement: The general licensee shall evaluate any changes to the written evaluations required by 10 CFR 72.212(b)(2) using the requirements of 10 CFR 72.48(c). A copy of this record shall be retained until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

Observation: Procedure 0.8, Section 1, Step 1.1, and Section 4, Step 4.3.1 required changes to be made to the 10 CFR 72 212 Evaluation Report in accordance with the provisions of 10 CFR 72.48 "Changes, Tests and Experiments." The current version of the 10 CFR 72.212 Evaluation Report was Revision 0. As such, no changes had been made and no 72.48 evaluations had been performed.

Retention requirement for the 10 CFR 72.212 Evaluation Report were established in Procedure 1.9, Step 2.6.4 which required the ISFSI dry fuel storage records to be maintained in accordance with 10 CFR Part 72. Further, the step required a retention period of five (5) years after the radioactive material was disposed of or transferred from the ISFSI. Procedure 1.9, Step 5.1.9 required that quality records generated in support of 10 CFR Part 72 be stamped as "ISFSI Records" and identified as "ISFSI" on the transmittal form (Attachment 1 of Procedure 1.9) from the document user to the document retention group. This stamp identified the category of record and the basis for the storage period in accordance with the plant Records Retention Schedule.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Administrative Procedure 0.8 "10CFR50.59 and 72.48 Reviews," Revision 18, (d) Administrative Procedure 0.29.1 "License Basis Document Changes," Revision 28, (e) Site Services Procedure 1.9 "Control and Retention of Records" Revision 50

Category: General License **Topic:** Seismic Acceleration

Reference: CoC 1004, Tech Spec 1.1.1.3 Amendment 9

Requirement: The site specific horizontal seismic acceleration level shall be 0.25g or less. The site specific vertical seismic acceleration level shall be 0.17g or less. This evaluation may be included in the 72.212(b) evaluation report.

Observation: The requirement of a maximum of 0.25g horizontal seismic acceleration and 0.17g vertical acceleration were met for the Cooper Nuclear Station ISFSI. The 10 CFR 72.212 Evaluation Report, Section 10.2.4 "Earthquake Intensity/Seismic Acceleration," discussed the earthquake potential at the ISFSI and referenced the site's Updated Safety Analysis Report (USAR), Table II-5-1 "Selected Design Earthquakes" which characterized the safe shutdown earthquake (SSE) for the site as 0.20g horizontal and approximately 0.13g vertical (2/3 of the horizontal component) at bedrock. Change Evaluation Document (CED) 6024681 "ISFSI Pad and Apron," provided a justification that the site bedrock spectra enveloped the ISFSI, canister, and horizontal storage module spectra.

The Cooper Nuclear Station site seismic design basis response was based on consideration of the "Taft" earthquake as part of the calculations of site structural

response and qualification. URS Corporation performed soil structure interaction analysis for the site and ISFSI pad area with resultant accelerations at the top of the pad using the Taft information. Black & Veatch performed the calculation for the pad and horizontal storage module response using input from URS. The seismic qualification of the horizontal storage module and the seismic acceleration limits noted in the NUHOMS Certificate of Compliance were based on Regulatory Guide 1.60 "Design Response Spectra for Seismic Design of Nuclear Power Plants." A comparison was required to ensure that the site ISFSI pad predicted maximum acceleration did not exceed the Certificate of Compliance technical specification limit and was below the horizontal storage module seismic qualification acceleration.

The Cooper Nuclear Station USAR, Section II-5.2.4 "Application of the Design Earthquake Criteria" stated that the combined stresses resulting from dead, live, pressure, thermal, and earthquake having a peak ground acceleration of 0.2g are applied to structures, systems and components that are necessary to achieve safe shutdown. The design values of the vertical component of the accelerations are two-thirds those of the horizontal component for structural design. In that the site seismic spectrum is enveloped by the Regulatory Guide 1.60 spectrum, the design basis requirements noted in the Certificate of Compliance are generically satisfied. However, using the peak ground acceleration of 0.25g horizontal and 0.17g vertical acceleration levels as design qualifiers for the site (and ISFSI pad support structure additions) would neglect the seismic comparisons of the spectral shape of the resulting site specific pad design earthquake at the site. The reason was the shape of Regulatory Guide 1.60 type response spectrum, which was used in the NUHOMS Updated Final Safety Analysis Report (UFSAR) to account for seismic amplification occurring between the top of the pad and the center of gravity of the horizontal storage module, did not characterize the spectral shape of the approved SSE ground motions at the Cooper Nuclear Station or include any ground composition modifications. The resulting calculated response spectra at the top of the pad must be used to show that the resulting acceleration at the center of gravity of the horizontal storage module were bounded by the limits specified in the NUHOMS UFSAR for the horizontal and vertical directions.

In order to establish the amplification factor associated with the generic design basis response spectra, various frequency analysis were performed by Transnuclear for the NUHOMS components. In particular, the seismic design for the HSM-202 horizontal storage module with a 61BT canister was consistent with the spectra in the NUHOMS UFSAR, Section 3.2.3 "Seismic Design Criteria," with the exception that Regulatory Guide 1.60 response spectra was anchored to a maximum ground acceleration of 0.30g (instead of 0.25g) for the horizontal components and 0.20g (instead of 0.17g) for the vertical component. This was based on results of the frequency analysis of the HSM-202 structure which yielded the lowest frequency of 23.2 Hz in the transverse direction and 28.4 Hz in the longitudinal direction. The lowest vertical frequency exceeded 33 Hz. Thus, based on the Regulatory Guide 1.60 response spectra amplifications, the corresponding seismic accelerations used for the design of the HSM-202 were 0.37g and 0.33g in the transverse and longitudinal directions respectively and 0.20g in the vertical direction.

Site and pad structural analysis for the ISFSI showed the maximum design acceleration

at the top of the pad basemat in the longitudinal direction was 0.35g, which was below the 0.37g maximum specified in the NUHOMS UFSAR, Section 3.2.3 and Appendix K center of gravity. The maximum acceleration in the transverse direction was 0.33g, which was below the 0.37g maximum specified in the UFSAR. Based on these various calculations, the licensee determined that the seismic acceleration requirements specified in Technical Specification 1.1.1.3 were met.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Earth Sciences (Teledyne) Report "Earthquake Analysis of the Reactor Building Cooper Nuclear Station," for Burns and Roe, Inc., dated 1968, (c) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (d) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24, (e) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (f) Change Evaluation Document CED 6024681 "ISFSI Pad and Apron," Revision 0, (g) Regulatory Guide 1.60 "Design Response Spectra for Seismic Design of Nuclear Power Plants." dated December 1973

Category: General License **Topic:** Site Average Temperatures

Reference: CoC 1004, Tech Spec 1.1.1.1 Amendment 9

Requirement: The maximum average yearly temperature with solar incidence shall be 70 degrees F or less. The average daily ambient temperature shall be 100 degrees F or less. This evaluation may be included in the 72.212(b) evaluation report.

Observation: The temperature averages specified in Technical Specification 1.1.1.1 bounded the temperatures at the Cooper site. For the average yearly temperature, Section 10.2.1.1 "Average Yearly Temperature" of the 10 CFR 72.212 Evaluation Report described the methodology used to verify that the technical specification limit was not exceeded. For the years 1995 through 2008, ambient temperature data from the site meteorological tower was obtained. For each month, a daily average temperature was determined. The average daily temperatures for each month in a given year were added together and divided by twelve to obtain the average yearly temperature. The highest average yearly temperature was 55 degrees F occurring in 2006. This value was less than the 70 degrees F maximum specified in Technical Specification 1.1.1.1. For the average daily temperature, Section 10.2.1.2 "Average Daily Ambient Temperature" of the 72.212 Evaluation Report describe how the temperature value was calculated. For each month, a daily average maximum temperature was determined. The highest daily average maximum temperature was 91.6 degrees F, occurring in July 2002. This value was less than 100 degrees F maximum.

For this inspection, these values were comparing with the Historical Climate Data Summaries from the High Plains Regional Climate Center Website at (http://www.hprcc.unl.edu/data/historical/) for the period 1990-2010 from the nearby city of Auburn, NE to confirm their validity. The annual monthly average temperature for Auburn in 2006 was 54.91 degrees F. For the period 1990-2010 the highest annual monthly average temperature at Auburn was 55.68 degrees F for the year 1998. The month with the highest monthly average maximum temperature for Auburn from 1990

through 2010 was 93.16 degrees F in July 2002. Therefore, the values determined by the licensee for the Cooper Nuclear Station site were reasonable for this area of the country.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) High Plains Regional Climate Center Website at http://www.hprcc.unl.edu/data/historical/

Category: General License **Topic:** Site Temperature Extremes

Reference: CoC 1004, Tech Spec 1.1.1.2 Amendment 9

Requirement: For HSMs containing 24P, 52B, and 61BT canisters, the site specific temperature extremes shall be minus 40 degrees F with no solar incidence and plus 125 degrees F with solar incidence. This evaluation may be included in the 72.212(b) evaluation report.

Observation: The site specific limit for temperature extremes of minus 40 degrees F, with no solar incidence, and plus 125 degrees F, with solar incidence, were met at the Cooper Nuclear Station site. Section 10.2.1.3 "Temperature Extremes" of the 10 CFR 72.212 Evaluation Report referenced the Cooper Updated Safety Analysis Report (USAR), Section II-3.1.1 "Temperature," which stated that the historical temperature had exceeded 110 degrees F on five occasions since record taking began in 1888. All five of those cases occurred between the years 1934 and 1939. For years 1995 through 2008, ambient temperature data from the site meteorological tower was obtained for each month. Absolute maximum and minimum temperatures were determined. The maximum temperature recorded was 105.6 degrees F during July 1995. The minimum temperature recorded was minus 16.1 degrees F during January 1996.

For this inspection, these values were compared with graphical maximum and minimum temperature data from the Historical Climate Data Summaries from the High Plains Regional Climate Center Website (http://www.hprcc.unl.edu/data/historical/) showing temperature daily extremes for the period 1893-2010 from the nearby city of Auburn, NE. The graphical data supported the licensee's data. At no time during that period did the daily extreme high exceed 120 degrees F for a maximum temperature or an extreme low of less than minus 40 degrees F for a minimum temperature.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Cooper Nuclear Station Updated Safety Analysis Report (USAR), Revision 24, (d) High Plains Regional Climate Center Website at http://www.hprcc.unl.edu/data/historical/

Category: General License **Topic:** Written Procedures Required

Reference: 10 CFR 72.212(b)(9) Published 2010

Requirement: The licensee shall conduct activities related to storage of spent fuel under this general license only in accordance with written procedures.

Observation: Written procedures were developed for the various handling, loading, movement,

surveillance, and maintenance activities associated with the dry cask storage activities. The procedures were consistent with the NUHOMS Updated Final Safety Analysis Report (UFSAR), Section 9.4.1 "Procedures," Section K.8.1 "Procedures for Loading the Cask," Section K.8.2 "Procedure for Unloading the Cask," and Section K.9 "Test and Maintenance." Procedures included a purpose, precautions and limitations, equipment and material needed, prerequisites to perform the work, procedural steps to perform the work, and acceptance criteria. The procedural steps were detailed and clear as to the action needed to perform the work with cautions provided in appropriate locations. Technical specification requirements were included in the procedural steps with sign-offs confirming that the requirement had been met. Implementation of the procedures was demonstrated during the NRC observed pre-operational tests and during the NRC observed loading of the first canister. However, during work activities associated with the second cask, procedures were not followed concerning the draining of the annulus gap. This resulted in an unintentional partial draining of the neutron shield on the transfer cask. This issue is discussed in these Inspector Notes under the Category: Operations and the Topic: Unintentional Draindown of Transfer Cask and resulted in the NRC issuing a non-cited violation (NCV) for failure to follow procedures.

Numerous procedures were developed for all the specific tasks related to dry cask storage. The procedures reviewed during this inspection included the following. Procedures for the selection of the spent fuel elements for loading and special nuclear material (SNM) accountability included Nuclear Performance Procedures 10.21, 10.26, 10.36 and 10.48. The primary procedures for handling, loading, and movement of the spent fuel to the ISFSI were Nuclear Performance Procedures 10.36.1, 10.37, 10.38, 10.39 and 10.40. Activities necessary for unloading a cask were described in Nuclear Performance Procedures 10.36.1, 10.37.1, 10.38.1, 10.39.1, and 10.40.1. For pre-operational inspections and activities to prepare for the loading of a cask, the primary procedures were Nuclear Performance Procedures 10.41, 10.42, 10.43, and 10.44. Responses to unexpected events were described in Nuclear Performance Procedure 10.51, Emergency Procedure 5.1HSM, and Emergency Plan Implementing Procedure 5.7.1. Surveillances were performed using Surveillance Procedures 6.HSM-TEMP.601, 6.LOG.601, 6.LOG.602, and 6.MISC.601. Procedures related to the crane, heavy loads, and the use of slings included Maintenance Procedures 7.1.8, 7.1.8.1, 7.1.10, 7.2.73, 7.2.76, and 7.6.1. Procedures related to quality assurance, licensing, and classifying items in accordance with their safety class included Site Services Procedure 1.4QA and Administrative Procedures 0.19, 0.19.1, 0.29.1, and 0.29.9. The procedure used to perform safety reviews was Administrative Procedure 0.8. The procedures related to records and record retention were Site Services Procedure 1.9 and Administrative Procedure 0.19. The procedure related to training was Administrative Procedure 0.17. Instrument calibration requirements were included in Administrative Procedures 0.37 and 0.38. The use of the hydrogen monitoring equipment was included in Chemistry Procedures 8.5.6 and 8.5.7. Health physics activities were covered in numerous plant health physics procedures that covered a wide range of normal radiation protection activates onsite. Vendor procedures used for welding included TriVis Procedures 06260-CNS-OPS-01 and 06260-CNS-SS-8-A-TN. Non-destructive testing procedures included TriVis Procedures 06260-CNS-QP-9.201, 06260-CNS-QP-9.202 and RRL NDT Consulting, Inc. Procedure TN 61BT/61BTH-HMSLD.

Procedure compliance by the workers was observed to be good during the loading of the first canister observed by the NRC. Procedures were readily available in the work area and were sometimes carried by the workers. Individuals that did not have their procedure in hand while they were performing a task were sometimes approached by the NRC inspector and asked for an explanation of the work activity they were performing. On each occasion, the individual was able to clearly discuss his work task consistent with the procedure and knew where the nearest procedure was located or where the cask loading supervisor was that had assigned him the particular task and had required him to report back upon completion. Procedural steps were being signed-off on a controlled copy maintained by the cask loading supervisor or the person performing the task.

Documents Reviewed:

(a) Nuclear Performance Procedure 10.21 "Special Nuclear Materials Control and Accountability Instructions," Revision 41, (b) Nuclear Performance Procedure 10.26 "Fuel Classification of CNS Spent Fuel for Dry Storage and DOE Disposition," Revision 0, (c) Nuclear Performance Procedure 10.36 "Fuel Bundle Selection Process for Loading NUHOMS 61BT DSC," Revision 2 and Revision 3, (d) Nuclear Performance Procedure 10.36.1 "Fuel Loading/Unloading of a DSC," Revision 1 and Revision 3, (e) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 0 and Revision 5, (f) Nuclear Performance Procedure 10.37.1 "Shielded Canister Unloading," Revision 0, (g) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4 and Revision 11, (h) Nuclear Performance Procedure 10.38.1 "Dry Shielded Canister Unsealing," Revision 1, (i) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 0, Revision 2, Revision 7 and Revision 8, (j) Nuclear Performance Procedure 10.39.1 "Dry Shielded Canister Transport from ISFSI to Reactor Building," Revision 0, (k) Nuclear Performance Procedure 10.40 "Dry Shielded Canister Transfer from Transfer Cask to HSM," Revision 4, (l) Nuclear Performance Procedure 10.40.1 "Dry Shielded Canister Transfer from HSM to Transfer Cask," Revision 0, (m) Nuclear Performance Procedure 10.41 "DSC Inspection and Pre-Operational Testing," Revision 0, (n) Nuclear Performance Procedure 10.42 "Transfer Trailer Inspection and Pre-Operational Testing," Revision 0, (o) Nuclear Performance Procedure 10.43 "Transfer Cask Offloading and Inspection," Revision 0, (p) Nuclear Performance Procedure 10.44 "Ancillary Equipment Procedure," Revision 0, (q) Nuclear Performance Procedure 10.48 "Caskworks Input Data Generation," Revision 1, (r) Nuclear Performance Procedure 10.51 "ISFSI/DFS Abnormal Operations," Revision 0 and Revision 1, (s) Emergency Procedure 5.1HSM "HSM Integrity," Revision 1, (t) Emergency Plan Implementing Procedure (EPIP) 5.7.1 "Emergency Classification," Revision 42, (u) Surveillance Procedure 6.HSM-TEMP.601 "HSM Thermal Performance Monitoring," Revision 0 and Revision 1, (v) Surveillance Procedure 6.LOG.601 "Daily Surveillance Log - Mode 1, 2, and 3," Revision 106, (w) Surveillance Procedure 6.LOG.602 "Daily Surveillance Log - Mode 4 or 5," Revision 53, (x) Surveillance Procedure 6.MISC.601 "Reactor Building Crane Inspection or Lift and Hold Operability Test for Cask Handling Operations," Revision 6 and Revision 10, (y) Maintenance Procedure 7.1.8 "Rigging and Lifting at CNS," Revision 26, (z) Maintenance Procedure 7.1.8.1 "Material Handling," Revision 2, (aa) Maintenance Procedure 7.1.10 "Qualification for Crane and Hoist Operators and Riggers," Revision 4, (bb) Maintenance Procedure 7.2.73 "Reactor, Turbine Building Crane Examination, Maintenance and Testing," Revision 14, (cc) Maintenance Procedure 7.2.76 "Sling, Fall Protection Harness/Lanyard Examination. Maintenance and Testing," Revision 8, (dd)

Maintenance Procedure 7.6.1 "Reactor Building Crane Operations," Revision 24, (ee) Site Services Procedure 1.4QA "Quality Assurance/Safety Classification Requirements," Revision 5, (ff) Site Services Procedure 1.9 "Control and Retention of Records," Revision 50, (gg) Administrative Procedure 0.8 "10CFR50.59 and 10CFR72.48 Reviews," Revision 18, (hh) Administrative Procedure 0.17 "Selection and Training of Station Personnel," Revision 54, (ii) Administrative Procedure 0.19 "Equipment Record and Functional Location File Program," Revision 25, (jj) Administrative Procedure 0.19.1 "Quality Assurance Program Applicable to Dry Fuel Storage," Revision 1, (kk) Administrative Procedure 0.29.1 "Licensing Basis Document Changes," Revision 28, (ll) Administrative Procedure 0.29.9 "ISFSI Licensing Basis Document Maintenance," Revision 0, (mm) Administrative Procedure 0.37 "Measuring and Test Equipment (M&TE) Calibration Program Guidelines," Revision 24, (nn) Administrative Procedure 0.38 "Processing Instrument Calibration Program," Revision 5, (oo) Chemistry Procedure 8.5.6 "Hy-Alerta Hydrogen Specific Detection Instrument," Revision 0, (pp) Chemistry Procedure 8.5.7 "Hy-Optima 1700 Hydrogen Specific Detection Instrument," Revision 0, (qq) Radiation Protection Procedure 9.ALARA.1 "Personnel Dosimetry and Occupational Radiation Exposure Program," Revision 38, (rr) TriVis Procedure 06260-CNS-OPS-01 "Spent Fuel Cask Welding: 61BT NUHOMS Canister," Revision 3, (ss) TriVis Welding Procedure Specification WPS 06260-CNS-SS-8-A-TN "Welding Procedure Specification," Revision 2, (tt) TriVis NDE Services, LLC Procedure 06260-CNS-QP-9.201 "Visual Weld Examination," Revision 6, (uu) TriVis NDE Services, LLC Procedure 06260-CNS-QP-9.202 "Color Contrast Liquid Penetrant (PT) Examination Using the Solvent Removable Method," Revision 6, (vv) RRL NDT Consulting, LLC Procedure TN 61BT/61BTH-HMSLD "Specific Procedure for HMSLD Leak Testing of Transnuclear NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel Inner Top Cover Plate and Vent and Siphon Port Cover Plates," Revision 0, (ww) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10

Category: Heavy Loads **Topic:** Heavy Lifts Outside the Spent Fuel Building

Reference: CoC 1004, Tech Spec 1.2.10 Amendment 9

Requirement: When handling a loaded transfer cask at a height greater than 80 inches outside the spent fuel building, a special lifting device that has at least twice the normal stress design factor for handling heavy loads, or a single failure proof handling system shall be used. In the event of a drop from a height greater than 15 inches, the fuel in the canister shall be returned to the spent fuel pool, the canister shall be removed from service, and the transfer cask shall be inspected for damage. The canister shall not be returned to service until a determination is made that it will continue to provide confinement. The transfer cask shall not be returned to service until a determination is made that it will continue to provide its design functions of transfer and shielding.

Observation: There were no lifts of the transfer cask performed outside the spent fuel building (reactor building). Therefore the special lifting device requirements were not applicable. The requirement related to dropping the cask from a height greater than 15 inches was included in Procedure 10.39, Step 6.7 which stated "If the loaded transfer cask drops greater than 15 inches at any time, stop and place equipment in a safe condition and refer to Technical Specification 1.2.10 Actions."

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 8

Category: Heavy Loads **Topic:** Inspections Prior to Each Use

Reference: UFSAR 1004, Section 4.5.1 Revision 10

Requirement: Visually inspect the following components prior to each use: a) the transfer cask exterior, trunnions and lifting yoke, for cracks, dents, gouges, tears, or damaged bearing surfaces; b) all threaded parts and bolts for burrs, chafing, distortion or other damage; c) all quick disconnect fittings for proper operation; d) the transfer cask interior surfaces for indications of excessive wear; and e) the neutron shield tank jacket for indications of damage.

Observation: Procedures involving the lifting of the transfer cask had incorporated the required inspection criteria. The procedures included 10.37, 10.37.1, 10.38, 10.39, and 10.39.1. Procedure 10.37 provided instructions for movement of an empty transfer cask and spent fuel canister from the transport trailer to the cask wash down area and then from the cask wash down area to the spent fuel pool. Procedure 10.38 provided instructions for the movement of the loaded transfer cask/spent fuel canister from the spent fuel pool to the cask wash down area. Procedure 10.39 provided instructions for the movement of a loaded transfer cask/spent fuel canister from the cask wash down area onto the trailer for transport to the ISFSI. Procedure 10.39.1 provided instructions for the movement of a loaded transfer cask/spent fuel canister from the trailer to the cask wash down area for unloading purposes. Procedure 10.37.1 provided instructions for the movement of a loaded transfer cask/spent fuel canister from the cask wash down area into the spent fuel pool for unloading purposes. Each of these procedures provided instructions to perform an inspection of the transfer cask prior to use. For example, Step 3.2.17 of Procedure 10.37 required the following inspections of the transfer cask prior to use: Step 3.2.17.1 required visual inspection of the cask exterior for cracks, dents, gouges, tears, or damaged bearing surfaces, stating particular attention should be paid to the cask trunnions and lift yoke; Step 3.2.17.2 required visual inspection of all threaded parts and bolts for burrs, chafing, distortion, or other damage; Step 3.2.17.3 required checking all quick-connect fittings to ensure their proper operation; Step 3.2.17.4 required visual inspection of the interior surface of the cask for any indications of excessive wear to bearing surfaces; and Step 3.2.17.5 required visual inspection of the neutron shield jacket for damage. All the other procedures listed above contained similar steps but in different locations in the respective procedure.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 0, (c) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (d) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 0, (e) Nuclear Performance Procedure 10.39.1 "Dry Shielded Canister Transport from ISFSI to Reactor Building," Revision 0, (f) Nuclear Performance Procedure 10.37.1 "Dry Shielded Canister Unloading," Revision 0

Category: Heavy Loads **Topic:** Procedures

Reference: NUREG 0612, Section 5.1.1 (2) Issued July 1980

Requirement: Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. The procedures should include: a) identification of the required equipment; b) inspections and acceptance criteria required before movement of the load; c) the steps and proper sequence to be followed in handling the load; d) defining the safe load path; and e) special precautions.

Observation: The procedures utilized at the Cooper Nuclear Station for the movement of the spent fuel canister contained the required elements listed above. The lifting operations that could occur over or in proximity to spent fuel or reactor safe shutdown equipment were governed by Procedures 10.37, 10.37.1, 10.38, 10.39, and 10.39.1. Procedure 10.37 governed movement of an empty transfer cask and canister from the transport trailer to the cask washdown area in the reactor building, then from the cask washdown area to the spent fuel pool. Procedure 10.38 governed the movement of the cask with a canister loaded with spent fuel from the spent fuel pool to the cask washdown area. Procedure 10.39 governed movement of the cask from the cask washdown area onto the transport trailer for transport to the ISFSI pad. Procedure 10.39.1 governed the movement of a cask from the transport trailer to the cask washdown area, for unloading purposes. Procedure 10.37.1 governed movement of a cask from the cask washdown area into the spent fuel pool, for unloading purposes. For example, the steps of Section 3.1 to Procedure 10.37 listed all the required equipment for the operation. The steps to Section 3.2.17 listed inspections that were required of the transfer cask prior to use. The steps of Sections 4.0 and 7.0 controlled the sequence of handling the load from the transport trailer to the cask washdown area and then from the cask washdown area to the spent fuel pool. Attachment 8 "Heavy Loads Path" contained a drawing of the safe load path that was to be followed. The safe load path was also provided in Procedure 10.38, Attachment 7 "Heavy Loads Path." Throughout the procedure, caution statements were utilized to draw attention to special precautions. All the other procedures listed above contained detailed steps related to the activities associated with the respective procedure.

Documents Reviewed: (a) NUREG 0612 "Control of Heavy Loads at Nuclear Power Plants," issued July 1980, (b) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 0, (c) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (d) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 2, (e) Nuclear Performance Procedure 10.39.1 "Dry Shielded Canister Transport from ISFSI to Reactor Building," Revision 0, (f) Nuclear Performance Procedure 10.37.1 "Dry Shielded Canister Unloading," Revision 0

Category: Heavy Loads **Topic:** Seismic Restraints

Reference: CoC 1004, Tech Spec 1.2.16 Amendment 9

Requirement: Seismic restraints shall be provided in the spent fuel building to prevent overturning of a loaded transfer cask if the horizontal acceleration at the transfer cask center of gravity is 0.40g or greater. The center of gravity horizontal acceleration calculation must be based on the site peak horizontal ground acceleration, but shall not exceed 0.25g.

Observation: Transfer cask seismic restraints were provided at the cask work station where welding,

drying, and sealing were performed. Calculation NEDC 09-065 addressed sliding and overturning of the loaded transfer cask, as well as restraint loading, during a safe shutdown earthquake (SSE). The licensee determined that the transfer cask would not overturn during a SSE while in the spent fuel pool due to hydraulic damping of the water. Seismic restraint loadings were calculated by analysis of the effects of the SSE accelerations in the reactor building floor.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Engineering Design Calculation (NEDC) 09-065 "Transfer Cask Tipping Evaluation"

Category: Heavy Loads **Topic:** Transfer Cask Alignment

Reference: CoC 1004, Tech Spec 1.2.9 Amendment 9

Requirement: Prior to canister insertion or retrieval, the transfer cask must be aligned to the HSM such that the longitudinal centerline of the canister is within 1/8 inch of its true position.

Observation: The technical specification alignment tolerance for the transfer cask to the horizontal storage module was specified in the licensee's procedure. Procedure 10.40, Section 7.0 "Final Alignment of Transfer Cask to HSM" specified the technical specification limit of ± 1/8 inch for the vertical and horizontal alignment tolerance. Step 7.5 required repeating the vertical and horizontal alignment checks to verify they were within ± 1/8 inch of their respective centerline targets prior to insertion. Steps 8.1 and 8.2 were quality control hold points that required a quality assurance individual to verify and sign that the transfer cask was properly aligned to within ± 1/8 inch of the vertical and horizontal centerline alignment targets before insertion. Step 8.2 referenced Technical Specification 1.2.9.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.40 "Dry Shielded Canister Transfer from the Transfer Cask to the HSM," Revision 4

Category: Heavy Loads **Topic:** Transfer Cask Lift Height Limits

Reference: CoC 1004, Tech Spec 1.2.13 Amendment 9

Requirement: A loaded transfer cask shall not be lifted inside the spent fuel building when canister basket temperature is below minus 20 degrees F. When canister basket temperature is between minus 20 degrees F and 0 degrees F, the transfer cask may be lifted to a maximum of 80 inches. When canister basket temperature is greater than 0 degrees F, no lifting height limits are imposed on the transfer cask. A loaded transfer cask shall not be lifted outside the spent fuel building when canister basket temperature is below zero degrees F.

Observation: The lift height limit requirement had been incorporated into Procedure 10.38, Step 2.11 which stated "No lifts or handling of a loaded transfer cask/canister at any height are permissible at canister basket temperatures below zero degrees F while inside the reactor building." Procedure 10.39, Step 2.8. stated "No lifts or handling of a loaded transfer cask/canister at any height are permissible at canister basket temperatures below zero

degrees F while inside the reactor building." A reference was made to Technical Specification 1.2.13. No lifts of the transfer cask take place outside the reactor building.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (c) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 0

Category: Heavy Loads **Topic:** Transfer Cask Operations in Direct Sunlight

Reference: CoC 1004, Tech Spec 1.2.14 Amendment 9

Requirement: Transfer operations shall not be conducted when the transfer cask is exposed to direct insolation at an ambient temperature of 100 degrees F or greater. When ambient temperatures exceed 100 degrees F, a solar shield shall be used to provide protection against direct solar radiation.

Observation: This requirement was incorporated into Procedure 10.39, Step 7.14 which stated "If outdoor ambient temperature is greater than 100 degrees F, or expected to be greater than 100 degrees F, install a solar cover."

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 0

Category: Heavy Loads **Topic:** Transfer Cask Trunnion Load Test

Reference: UFSAR 1004, Sections 4.2.3.3 Revision 10

Requirement: For transfer casks fabricated under the general license, a one-time pre-service load test of the trunnions is performed at a load equal to 150% of the design load followed by an examination of all accessible trunnion welds. Neither the transfer cask nor the trunnions are special lifting devices per ANSI N14.6

Observation: The NUHOMS Updated Final Safety Analysis Report (UFSAR), Section 4.2.3.3 "Onsite Transfer Cask" stated that the transfer cask was not an American National Standards Institute (ANSI) N14.6 special lifting device and only required a 150% load test of the trunnions. Section 4.2.3.3 also stated that the upper lifting trunnions and trunnion sleeves were conservatively designed in accordance with ANSI N14.6 stress allowable requirements for a non-redundant lifting device. Technical Specification 1.1.4 "Heavy Loads Requirements" stated that the transfer cask design had been reviewed and found to meet ANSI N14.6. Testing requirements in ANSI N14.6, Section 6.3 "Testing" required a 300% load test for special lifting devices. Though not required by the UFSAR, a 300% load test was completed on the transfer cask. The OS197H was rated at 125 tons as stated in UFSAR Section 1.3.2.1 "Onsite Transfer Cask." The upper trunnions were load tested to 750,000 lbs (375 tons) by Equipos Nucleares S.A. (ENSA) on April 6, 2002 per ENSA Procedure 0FW9 CS 020. The load was applied for a period of ten minutes. After the load test, a liquid penetrate test was performed on each trunnions weld surfaces. No reportable indications were found.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Calculation Number 08-042 "Transnuclear Transfer Cask (TC) and Dry Shielded Canister (DSC) Weights for Various Spent Nuclear Fuel Loading Configurations," Revision 0, (c) Equipos Nucleares S.A. (ENSA) Document 0FW9 CS 020 "Load Test of Upper Trunnions," Revision 2, (d) ENSA PT 1FW9/28 "Liquid Penetrate Certification," dated April 6, 2002, (e) American National Standards Institute N14.6 "Special Lifting Devices for Shipping Containers Weighing 10,000 lbs or More," Revision 1986

Category: Non-Destructive Exam **Topic:** Developer Drying Time

Reference: ASME Section V, Article 6, T-676.1 Code Year 2001

Requirement: Final interpretation shall be made after allowing the penetrant to bleed-out for 7-60 minutes under standard temperatures (50 and 125 degrees F).

Observation: Procedure QP 06260-CNS-QP-9.202, Section 6.2 "Application of Penetrant" specified application "dwell" times consistent with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section V, Article 6, T-676.1 requirement. In addition, dwell times were provided for higher temperatures. Section 6.2 provided a table for dwell times versus surface temperature for the Sherwin K017 penetrant. For temperatures of 50 to 75 degrees F, the dwell time was 30 minutes. For temperatures of 76 to 125 degrees F, the dwell time was 10 minutes. For temperature of 126 to 200 degrees F, the dwell time was 3 minutes. For temperatures of 201 to 325 degrees F, the dwell time was 1 minute. During the examination of the root pass weld on the outer lid of the first canister loaded, discussions with the NDE examiner concerning dwell times demonstrated she knew the correct dwell times from memory.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section V "Nondestructive Examinations," 2001 Edition, (b) TriVis Procedure QP 06260-CNS-QP-9.202 "Color Contrast Liquid Penetrant (PT) Examinations Using the Solvent Removable Process," Revision 6

Category: Non-Destructive Exam **Topic:** Helium Leak Rate

Reference: CoC 1004 Tech Spec 1.2.4a Amendment 9

Requirement: The helium leak rate on the inner top cover seal weld shall be less than or equal to 1.0 x 10(-7) reference cubic centimeters/second.

Observation: The first canister loaded met the technical specification requirements for the helium leak rate on the inner top cover seal with a leak rate of 5.6 x 10(-10) standard (std) cubic cm (cc)/sec helium. Procedure TN 61BT/61BTH-HMSLD was used to perform the helium leak rate test on the inner lid. Section 2.2 of the procedure specified an acceptance criteria of less than or equal to 1.0 x 10(-7) reference cc/sec for air leakage. Section 2.2 of the procedure converted the air leak rate to a helium leak rate of 2.0 x 10(-7) std cc/sec. This helium leak rate limit was consistent with the definition of leak tight in the American National Standards Institute (ANSI) guidance N14.5-1997, Section 2.1 and reflected the conversion from reference cc/sec of air to standard cc/sec for helium. A note to the definition in ANSI N14.5-1997 stated that 1 x 10(-7) ref-cc/sec of air was equivalent to 2 x 10(-7) cc/sec of helium. Section 4.1 of Procedure TN 61BT/61BTH-

HMSLD required a test system with a minimum leak rate sensitivity of 1.0 x 10(-9) std cc/sec. The leak detector used for the inner top cover seal leak test had a minimum leak rate sensitivity of 1.0 x 10(-10) std cc/sec. Section 3.0 required the person performing the examination to be a Level II or Level III examiner qualified and certified in helium mass spectrometer leak detection (HMSLD) in accordance with Recommended Practice SNT-TC-1A. The individual performing the leak testing on the inner lid and the vent and siphon port covers for the first canister observed by the NRC was a qualified Level II examiner.

Helium mass spectrometer leak detector, Model # UL200, Serial # 20028609 was used for the test. The calibration due date was August 6, 2011. The minimum sensitivity of the detector was 1.0 x 10(-10) std cc/sec. A calibrated leak source was used to verify proper calibration of the leak detector before use. Calibrated leak source, Model GPP-7-He-QF25-110CC, Serial # 5029 was used to verify proper operations of the leak detector. The calibrated helium leak source had a leak rate of 1.64 x 10(-7) atmosphere-cc/sec at 22.9 degrees C and had been calibrated January 22, 2010 with a calibration due date of January 22, 2011. The temperature coefficient was 4% per degrees C. Correcting the leak rate for temperature yielded a helium leak rate for the calibrated source of 1.9 x 10(-7) std cc/sec. The reading on the leak detector when the calibrated leak source was open, taken prior to the leak test for the inner lid, was 1.9 x 10(-7) std cc/sec and 2.1 x 10(-10) std cc/sec when the leak source was closed. The leak rate on the inner lid seal to demonstrate compliance with Technical Specification 1.2.4a was performed after the outer top cover plate root pass weld was completed (Procedure 10.38, Step 13.12). The helium leak detector was connected to the outer top cover plate test plug (Step 13.10). The canister had been filled with helium to 2.443 psig in compliance with Technical Specification 1.2.3a. (Step 11.10). Testing the volume in the gap between the inner lid and the outer lid effectively verified the integrity of the welds on both the inner lid and the welds on the vent and siphon port covers and provided for a cumulative, quantitative leakage rate value for the canister closure containment boundary.

As an additional verification to confirm the integrity of the vent and siphon port covers, the licensee had performed an informational helium leak test of the two port covers after welding of the covers was completed and before the outer top cover plate was installed. This leak test was performed per Procedure 10.38, Step 12.1.5 using Procedure TN 61BT/61BTH-HMSLD. The acceptance criteria for this test was 3.4 x 10(-8) std cc/sec per Step 9.1 of Procedure TN 61BT/61BTH-HMSLD. For the first canister, the leak rate on the vent port was 8.6 x 10(-10) std-cc/sec. The leak rate on the siphon cover was 8.3 x 10(-10) std-cc/sec.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) RRL.NDT Consulting Procedure TN 61BT/61BTH-HMSLD "Specific Procedure for HMSLD Leak Testing of Transnuclear NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel Inner Top Cover Plate and Vent and Siphon Port Cover Plates," Revision 0, (c) VTI Certificate of Calibration Report Number 5279-ACAL-COMP-1-48405 for Calibrated Leak Source Model GPP-7-He-QF25-110CC, Serial # 5279, (d) Leak Test Examination Certificate and Visual Acuity Examination Record for Robert Kyle Limoge, dated September 27, 2010, (e)

American National Standards Institute (ANSI) N14.5 "Leakage Tests on Packages for Shipment," dated 1997, (f) Recommended Practice SNT-TC-1A "Society for Nondestructive Testing-Technical Council," dated 1992

Category: Non-Destructive Exam **Topic:** Liquid Penetrant Testing

Reference: CoC 1004, Tech Spec 1.2.5 Amendment 9

Requirement: Welds on the inner and outer top cover shall be dye penetrant tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Article NB-5000. The liquid penetrant test acceptance standards shall be those described in Subsection NB-5350 of the Code.

Observation: The liquid penetrant exam requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Article NB-5000, Subsection NB-5350 "Liquid Penetrant Acceptance Standards" were incorporated into Procedure 06260-CNS-QP-9.202 for the outer and inner top cover. Section 7.1 of the procedure specified the acceptance criteria for the welds. This acceptance criteria was identical to the criteria in Subsection NB-5352 of the code. Procedure 06260-CNS-OPS-01, Attachment 9.2 "Weld Map" specified in Note #8 that liquid penetrant exams were to be performed in accordance with Subsection NB-5350 of the ASME code. Attachment 9.3 "Weld Data Sheet" provided a list of the required liquid penetrant tests (PT). For the inner top cover to shell welds, PT was required on the tack welds, root weld, and final weld. For the siphon port and vent port, PT was required on the tack welds, root weld, and final weld. For the outer top cover to shell welds, PT was required on the tack welds, root weld, intermediate weld, and final weld. For the test plug welds, PT was required on the initial plug weld, root weld, and final weld. Procedure 06260-CNS-QP-9.202 was qualified for use with temperature ranges from 50 degrees F to 325 degrees F. High temperature liquid penetrant was used for the exam on the first canister. The temperature on the lid was 130 degrees F. The NRC inspectors observed the liquid penetrant examinations of the first canister. No indications were found on the welds for the inner lid, outer lid, vent and siphon ports, and the test plug that required repair on the first canister loaded.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) TriVis Procedure 06260-CNS-OPS-01 "Spent Fuel Cask Welding - 61BT NUHOMS Canister," Revision 3, (c) TriVis Procedure 06260-CNS-QP-9.202 "Color Contrast Liquid Penetrant (PT) Examination Using the Solvent Removal Method," Revision 6, (d) TriVis NDE Personnel Qualification Summary for Greg Miaris (including Certification Record and Visual Acuity Record), dated May 23, 2009

Category: Non-Destructive Exam **Topic:** Permanent Record

Reference: ASME Section V, Article 6, T-676 Code Year 2001

Requirement: The inspection process, including findings (indications) shall be made a permanent part of the user's record by video, photograph or other means which provide an equivalent record of weld integrity. The video or photographic record should be taken during the

final interpretation period.

Observation: Procedure 06260-CNS-OPS-01, Attachment 9.5 "Field Comments and/or Repair Log" provided a section for sketching and documenting repair details, should this be necessary. No records were required for the welds on the inner and outer lid of the first canister, since no indications were found on the welds and no repairs were required.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section V "Nondestructive Examinations," 2001 Edition, (b) TriVis Procedure 06260-CNS-OPS-01 "Spent Fuel Cask Welding 61BT NUHOMS Canister," Revision 3

Category: Non-Destructive Exam **Topic:** Unacceptable Fusion

Reference: ASME Section III, Article NF-5360 Code Year 2001

Requirement: For fillet welds, incomplete fusion of more than 3/8 inch (10 mm) in any 4 inch (102 mm) segment or incomplete fusion of more than 1/4 inch (6 mm) in welds less than 4 inches is unacceptable. For groove welds, any incomplete fusion is unacceptable. Rounded end conditions (start and stop) shall not be considered indications of incomplete fusion.

Observation: Procedure 06260-CNS-QP-9.201, Step 6.3.B stated that any incomplete fusion on the lid welds was unacceptable. No unacceptable fusion was found on the inner and outer lid welds on the first canister welded.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III "Rules for Construction of Nuclear Facility Components," 2001 Edition, (b) TriVis Quality Procedure 06260-CNS-QP-9.201 "Visual Weld Examination," Revision 6

Category: Non-Destructive Exam **Topic:** Unacceptable Indications

Reference: ASME Section III, Article NB-5352 Code Year 2001

Requirement: Only indications with major dimensions greater than 1/16 inch should be considered relevant. The following relevant indications are unacceptable: (1) any cracks or linear indications. Linear indications having a length at least 3 times greater than the width; (2) rounded indications with dimensions greater than 3/16 inch (4.8 mm); (3) more than four rounded indications in a line, separated by 1/16 inch (1.6 mm) or less edge to edge; and (4) more than ten rounded indications in any 6 square inch area in the most unfavorable location relative to the indications being evaluated.

Observation: All the requirements specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Article NB-5352 "Acceptance Standard" had been incorporated into Procedure 06260-CNS-QP-9.202, Section 7.1 "Acceptance Criteria." No unacceptable indications were found during the welding of the first canister.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III "Rules for Construction of Nuclear Facility Components," 2001 Edition, (b) TriVis Procedure QP 06260-CNS-QP-9.202 "Color Contrast Liquid Penetrant (PT) Examinations Using the Solvent Removable Process," Revision 6

Category: Non-Destructive Exam **Topic:** Unacceptable Undercut

Reference: ASME Section III, Article NF-5360 Code Year 2001

Requirement: Undercuts deeper than 1/32 inch (0.8 mm) on one side for the full length of the weld are unacceptable. Undercuts deeper than 1/32 inch (0.8 mm) on one side for one half the length of the weld and deeper than 1/16 inch (1.6 mm) on the same side for one-fourth the length of the weld are unacceptable.

Observation: The requirement from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Article NF-5360 had been incorporated into Procedure 06260-CNS-QP-9.201. Section 6.3 "Weld and HAZ Acceptance Criteria" of the procedure stated "Undercut (wash, scalloping or other related nonstandard terms) shall not exceed 1/32 inch in any direction. Welding of the first canister inner and outer lids did not have any unacceptable undercuts.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III "Rules for Construction of Nuclear Facility Components," 2001 Edition, (b) TriVis Quality Procedure 06260-CNS-QP-9.201 "Visual Weld Examination," Revision 6

Category: Operations **Topic:** Canister (DSC) Dewater

Reference: UFSAR 1004, Section K.8.1.3.18 Revision 10

Requirement: Dewater the canister using compressed air, nitrogen or helium introduced through the vent port to force the water from the canister cavity through the siphon port, prior to starting the vacuum drying process.

Observation: Dewatering of the canister was accomplished using helium connected to the vent port and a pump connected to the siphon port. Procedure 10.38, Section 5.0 "DSC Initial Pump-Down" removed 1100 gallons of water prior to welding. This was necessary to reduce the amount of water in the canister that could generate hydrogen. The 1100 gallons was removed by connecting a pump to the siphon port which drained into the spent fuel pool. Helium bottles were connected to the vent port. This arrangement was shown in Attachment 5 "DSC Pump-Down System" of Procedure 10.38 and resulted in helium replacing the water as it was pumped from the canister. During the pump-down process, approximately 1 to 5 psig helium pressure was maintained in the canister. Once the 1100 gallons was removed, the siphon port valve was closed and a slight helium overpressure was maintained. In preparations for welding the inner cover lid per Section 6 "Welding DSC Inner Top Cover," a helium supply was connected to the siphon port (Step 6.3.3), after which the helium supply to the vent port was disconnected (Step 6.3.3.1). This allowed the hydrogen monitor to be connected to the vent port to monitor hydrogen generation during the welding process. After the inner lid welding was complete, the helium continued to flow through the siphon line and out the vent port (Step 6.10). The helium supply was temporarily removed and the ports closed while a fit check of the outer top lid was performed (Steps 6.18 thru 6.21). The outer top lid was then removed and the helium supply and the pump and drain line were reconnected in accordance with Attachment 5 and the remaining water, approximately 500 gallons, was pumped out of the canisters through the siphon line as the helium, at a pressure of approximately 5 psig, was supplied to the canister through the vent line. The removal of the remaining water was performed using Section 7 "DSC Final Pump-Down." Once no

more water was observed coming out of the drain line, the canister was isolated and the vacuum drying system connected to the canister.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4

Category: Operations **Topic:** Canister (DSC) Preparation

Reference: UFSAR 1004, Section 1.3.3 Revision 10

Requirement: The internals and externals of the canister are thoroughly washed or wiped down to ensure that the DSC will meet existing plant cleanliness requirements for placement into the spent fuel pool.

Observation: The licensee's operations procedure included the requirement for cleaning the canister prior to placement in the pool. Procedure 10.41, Step 4.3.3 stated "If dirt or debris is found in a fuel cell(s) the dry shielded canister (DSC) may require movement to an area to flush the canister with a pressure washer and demineralized water." During canister upending, Step 5.2 instructed the operators to wipe down the outside of the canister shell.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Nuclear Performance Procedure 10.41 "Dry Shielded Canister Inspection and Pre-operational Testing," Revision 0

Category: Operations **Topic:** First Cask Loading Completed

Reference: N/A

Requirement: The following provides information related to the first cask loaded at the Cooper Nuclear Station.

Observation: The first NUHOMS 61BT canister was loaded at the Cooper Nuclear Station and placed in the horizontal storage module (HSM) on October 21, 2010. The canister was loaded in accordance with the Transnuclear NUHOMS Certificate of Compliance (CoC) No. 1004, Amendment 9 and the NUHOMS Updated Final Safety Analysis Report (UFSAR), Revision 10. The 61BT canister was described in the UFSAR in Appendix K "NUHOMS 61BT System." The horizontal storage module used at Cooper was the HSM-202 and was described in Appendix V "NUHOMS HSM Model 202." The transfer cask used at Cooper was the OS197H. The OS197H transfer cask description was added to the UFSAR in Amendment 6. Canister # CNS61B-007-A was loaded with 61 intact BWR fuel elements with a total heat load of 11.3256 kW. The highest individual fuel element heat load was 231.6 watts. The canister was placed in the spent fuel pool and fuel loading initiated on October 13, 2010. Loading, welding, vacuum drying and helium backfill were completed and the canister inserted into HSM # DFS-HSMA-1A on October 21, 2010. A total of 1480 man-hrs was required to load the first cask with an estimated 0.7 person-rem.

Documents Reviewed: (a) Letter (NLS2010099) from David W. Van Der Kamp, Nebraska Public Power District to NRC Document Control Desk entitled "Thirty-Day Notification Pursuant to

10CFR72.212 Conditions of General license Issued Under 72.210 for Storage of Spent Fuel," dated November 15, 2010 [NRC ADAMS Accession No. ML103270518], (b) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10, (c) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9

Category: Operations **Topic:** First Systems Placed in Service

Reference: CoC 1004, Tech Spec 1.1.7; 10 CFR 72.4 Amendment 9

Requirement: The thermal performance of the first HSM placed in service shall be assessed. Thereafter, the thermal performance of HSMs with successively higher decay heat loads shall be assessed, up to the maximum canister heat load allowed under the Certificate of Compliance. The 61B canister is limited to 18.3 kW. A letter report summarizing the results of the assessment shall be submitted to the NRC within 30 days of placing the canister in the HSM.

Observation: The thermal performance of the first canister placed in service was assessed and a letter submitted to the NRC dated November 15, 2010, in compliance with Technical Specification 1.1.7. The first canister loaded was Canister Serial # CNS61B-007-A. The canister had a heat load of 11.3256 kW and was inserted into Horizontal Storage Model Serial # DFS-HSMA-1A on October 21, 2010. The horizontal storage models used at Cooper were HSM Model 202. The projected decay heat loads for the remaining seven canisters planned for loading in this first campaign were lower heat loads. Calculations to establish the maximum acceptable horizontal storage module air temperature rise (delta T) as a function of heat load and ambient temperature were performed in Transnuclear Calculation NUH004-0433 "Air Flow Calculation for NUHOMS HSM Model 202 with 61BT DSC," Revision 1. The calculations used the methodology required by Technical Specification 1.2.8 "HSM Maximum Air Exit Temperature with a Loaded 24P, 52B, 61BT, 32PT, 24PHB, or 24PTH-S-LC Only." The specified temperature rise was selected to ensure the fuel clad and concrete temperatures were maintained at or below acceptable long-term storage limits.

Thermal performance testing was conducted as described by Technical Specification 1.2.8. Daily inlet air temperature (ambient) and horizontal storage module outlet air temperature measurements were performed until thermal equilibrium was reached. Throughout the 11 day period following initial loading, the actual delta T values remained below the calculated delta T limits of 32 to 34 degrees F. Using a three day rolling average value, the thermal performance indicated an equilibrium was reached by November 1, 2010, eleven days after placement in the horizontal storage module. Delta T values between the ambient temperature and the horizontal storage module outlet air vent varied from 6.5 degrees F to 31 degrees F with considerable variance from day to day. Ambient temperatures had varied from 35.1 degrees F to 64.1 degrees F during the test period. The horizontal storage module exhaust values did not vary as widely during the test period going from a low of 61.9 degrees F to a high of 79.7 degrees F. The high ambient temperatures vs low ambient temperatures did not correlate with the high and low horizontal storage module exhaust temperatures.

The licensee made several observations about the variances seen in the temperature levels during the testing period and determined that changes in ambient temperatures and wind direction affected the horizontal storage module exhaust values. The delta T values showed higher step increases on days with a large drop in ambient temperature which was attributed to the release of additional latent heat stored in the concrete. For example, the October 28, 2010 ambient temperature measured 35.1 degrees F compared to the previous day of 53.9 degrees F (difference of 18.8 degrees F). The delta T was 27.3 degrees for October 28, 2010 compared to the previous day of 17.6 degrees. Over the next two days the temperature increased to 53.5 degrees F. Then the following day, on October 31, 2010, the temperature dropped to 44.5 degrees F, resulting in a delta T of 31 degrees F. This delta T related to a 9 degrees temperature drop, yet it produced a higher delta T than the October 28, 2010 delta T of 27.3 degrees F. Variations were also noted on October 23, 2010 (delta T = 9.4 degrees F) and October 26, 2010 (delta T = 6.5 degrees F) which were measured during high wind conditions. High wind conditions resulted in lower measured delta temperatures due to increased cooling from the additional convective heat transfer. The three day rolling average provided data that smoothed-out the swings in the daily values and provided a better understanding of the thermal conditions being experienced in the horizontal storage module.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (c) ISFSI HSM Temporary Temperature Monitoring Work Order #4774953 (CED 6024683), (d) Surveillance Procedure 6.HSM-TEMP.601, Revision 0, (e) Letter (NLS2010099) from David W. Van Der Kamp, Nebraska Public Power District to NRC Document Control Desk entitled "Thirty-Day Notification Pursuant to 10CFR72.212 Conditions of General license Issued Under 72.210 for Storage of Spent Fuel," dated November 15, 2010 [NRC ADAMS Accession No.103270518]

Category: Operations **Topic:** HSM Daily Thermal Monitoring

Reference: CoC 1004, Tech Specs 1.3.1, 1.3.2 Amendment 9

Requirement: HSM thermal performance may be monitored using Technical Specification 1.3.1 or 1.3.2. Technical Specification 1.3.1 requires a daily inspection of the HSM air inlets and outlets to ensure that no material accumulates between the modules that could block air flow. Technical Specification 1.3.2 requires a daily check of the HSM temperatures to identify off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. If air inlet and outlet temperatures are used, they must reflect the thermal performance of each individual module and not the combined performance of adjacent modules.

Observation: The licensee was implementing Technical Specification 1.3.2 by taking daily temperature readings for each of the horizontal storage modules. Surveillance Procedure 6.LOG.601, Attachment 21 "HSM Daily Thermal Performance Checks," or Surveillance Procedure 6.LOG.602, Attachment 10 "HSM Daily Thermal Performance Checks," were used to document the temperature readings depending on which mode of operation the reactor was in. In the event no temperature monitoring was available, both procedures directed that ISFSI Technical Specification 1.3.1 be entered concurrently and operators

were to visually inspect the horizontal storage module inlet and outlet vents daily.

The "Basis" statement for both Technical Specification 1.3.1 and 1.3.2 discussed limits on the horizontal storage module concrete temperature. Technical Specification 1.3.1 stated that the objective of the daily vent inspections was to ensure the concrete temperatures did not exceed 350 degrees F. The concrete temperature could increase to over 350 degree F in the accident circumstances of complete blockage of all vents if the blockage exceeded approximately 40 hours. If blockage of the vent was found during the daily inspection, the blockage was required to be cleared. Technical Specification 1.3.2 discussed taking temperature measurements that could show unexplained differences indicating problems related to the concrete or fuel clad temperature limits. Technical Specification 1.3.2 stated that if concrete temperatures exceeded 350 degrees F for more than 24 hours, the horizontal storage module must be removed from service unless it could be demonstrated that the structural strength of the horizontal storage module had an adequate margin of safety. Both Attachments 10 and 21 to the two surveillance procedures established limits on the horizontal storage module temperature and the daily heat-up rate based on Transnuclear Specification NUH-03-10102 in order to limit the horizontal storage module temperature rates to below the technical specification limits. The maximum limit for daily temperature was 217 degrees F with a lower administrative limit of 195 degrees F. An administrative limit for daily temperature rise was established as less than or equal to 11 degrees F. If the administrative limits were exceeded, Emergency Procedure 5.1HSM was initiated, Technical Specification 1.3.1 was entered, and the horizontal storage module vents were inspected daily. Emergency Procedure 5.1HSM required hourly temperature monitoring. If the temperature exceeded 217 degrees F for more than 24 hours, Technical Specification 1.3.2 was entered.

When a canister was loaded into a horizontal storage module, the daily temperature readings began. HSM-4A was loaded on December 3, 2010. It had a heat load of 11.2675 kW. The ambient temperatures experienced a large swing during the temperature monitoring of HSM-4A. Initially, the inlet temperature was 51 degrees F. The temperature dropped to the 20s and 30s during the majority of the monitoring period, reaching 12 degrees F on December 12, 2010. HSM-4A did not meet the required temperature acceptance criteria of Procedure 6.HSM-TEMP.601 on two occasions. On December 7 when the ambient temperature was 25 degrees F and the exhaust temperature was 58.1 degrees F, the delta T was 33.1 degrees. Procedure 6.HSM-TEMP.601, Attachment 3 "HSM Air Temperature Rise Table" listed the acceptable delta T as 31 degrees based on the ambient temperature and the heat load of the canister. On December 10, 2010, the ambient temperature was 34 degrees F and the exhaust temperature was 65.8 degrees F, for a delta T of 31.8 degrees. This was 0.8 degrees above the acceptable limit of 31 degrees F from Attachment 3. The failure to meet the delta T limit for HSM-4A was documented in Condition Report CR-CNS-2010-09089.

Horizontal storage module HSM 1B was loaded on December 10, 2010 at 1225 and started with a temperature reading of 32.5 degrees F. The following day at 0940 the temperature reading was 45.1 degrees F. The difference between the two readings was 12.6 degrees, which exceeded the 11 degrees heat-up limit within 24 hours in accordance with Procedure 6.LOG.601. The procedure had established the 11 degrees as an

administrative limit to maintain compliance with Technical Specification 1.3.1. Condition Report CR-CNS-2010-09222 was initiated identifying that the 11 degrees F limit had been exceeded and Emergency Procedure 5.1HSM was entered. No blockage of the vents were found and the 11 degrees administrative limit was not exceeded during the following daily readings. The problem was attributed to the faster heat-up that occurred during the initial 24 hr period after insertion when the horizontal storage module started out at a cold temperature. The canister had a heat load of 11.26 kW.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Surveillance Procedure 6.LOG.601 "Daily Surveillance Log - Modes 1, 2, and 3," Revision 106, (c) Surveillance Procedure 6.LOG.602 "Daily Surveillance Log - Modes 4 or 5," Revision 53, (d) Surveillance Procedure 6.HSM-TEMP.601 "HSM Thermal Performance Monitoring," Revision 1, (e) Emergency Procedure 5.1HSM "HSM Integrity," Revision 1, (f) Transnuclear Specification NUH-03-10102 "Specification for Generic Temperature Monitoring of NUHOMS HSM Model 202 Loaded with a DSC Containing 24 kW Total Maximum Decay Heat Load," Revision 1, (g) Condition Report CR-CNS-2010-09089 "HSM 1A Delta Temperature Exceeded," initiated December 7, 2010, (h) Condition Report CR-CNS-2010-09222 "HSM 1B Delta Temperature Exceeded," dated December 11, 2010

Category: Operations **Topic:** HSM Startup Monitoring - All Canisters

Reference: CoC 1004, Tech Spec 1.2.8 Amendment 9

Requirement: The equilibrium air temperature rise across an HSM containing any canister with a 24.0 kW decay heat load and spent fuel that has cooled for 5 years or more, shall not exceed 100 degrees F. For decay heat loads less than 24.0 kW, the equilibrium air rise must be calculated using the same methodology and inputs documented in the FSAR. If the actual temperature rise at equilibrium is less than 100 degrees F for the 24.0 kW loading or less than the calculated rise for a lower kW loading, no further startup thermal monitoring is required. The air temperature rise across the HSM shall be recorded at 24 hours following loading and daily thereafter until thermal equilibrium is reached. The air temperature must be measured in such a manner as to obtain representative values of inlet and outlet temperatures.

Observation: Surveillance Procedure 6.HSM-TEMP.601 provided instructions for conducting the horizontal storage module thermal performance monitoring by obtaining and recording temperatures at the horizontal storage module inlet and outlet vent per Technical Specification 1.2.8. Step 3.0 of Procedure 6.HSM-TEMP.601 provided direction for the setup and operate of the Fluke Model 51 Digital Thermometer to obtain the horizontal storage module temperatures. The temperature data was recorded in Attachment 1 "ISFSI Tech Spec 1.2.8 Data". Step 3.16 stated: "After loading the horizontal storage module, and within 24 hours, temperature rise shall be measured daily until equilibrium temperature is reached." Attachment 3 "HSM Air Temperature Rise Table" and Attachment 4 "HSM Air Temperature Rise Graph" provided acceptable temperature rise values for varying canister heat loads and ambient temperatures. Procedure 6.HSM-TEMP.601, Section 5.0 "Acceptance Criteria" stated that if the values recorded in Attachment 1 exceeded the acceptable temperature limits, then the action statement in Technical Specification 1.2.8 should be entered, the vents checked for blockage, and a

condition report issued. Attachment 7.0 "Sign-off and Review Sheet" required a signature from a Shift Manager that the acceptance criteria was met.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Surveillance Procedure 6.HSM-TEMP.601 "HSM Thermal Performance Monitoring," Revision 1

Category: Operations **Topic:** Hydrogen Monitoring

Reference: UFSAR 1004, Sections K.8.1.3.12 Revision 10

Requirement: Provide for continuous hydrogen monitoring of the DSC cavity during all lid cutting and welding operations to ensure that a safety limit of 2.4% is not exceeded. Purge with 2-3 psig helium (or other inert medium) as necessary to maintain the hydrogen concentration safely below this limit.

Observation: Continuous monitoring for hydrogen was required by procedures and performed during the loading of the first canister consistent with the requirements in the NUHOMS Updated Final Safety Analysis Report (UFSAR). Procedure 10.38, Steps 6.3.6 thru 6.3.8 stated "Connect hydrogen monitoring system to vent purge fitting, route vent line to sump or HEPA per radiation protection's direction and place hydrogen monitoring system in service." Step 6.3.10 required the hydrogen concentration in the canister to be below 2.4% prior to the start of welding. A note following Step 6.3.11 provided a caution statement which stated "Continuous monitoring of the DSC (canister) cavity for hydrogen gas is required during inner top cover plate welding. In addition, the inner cover plate weld joint shall be checked for hydrogen gas prior to the start of welding activities, including grinding. If the hydrogen concentration exceeds 2.4% (60% lower explosive limit), stop all welding operations until the hydrogen concentration decreases below 2.4%. Procedure 06260-CNS-OPS-01, which provided the detailed instructions for making the welds on the canister using the automatic welding system, included several cautionary statements directing the welding personnel to check for hydrogen in accordance with plant procedures before beginning spark producing work.

To further reduce the hydrogen levels that could build-up inside the canister, 1100 gallons of water were removed prior to welding (Step 5.4) and a helium purge was initiated through the siphon port during welding (Step 6.3.9) at a pressure of approximately 1 psig. Prior to the start of the root pass weld on the inner lid, a hand held hydrogen monitor (HY-ALERTA Model 500) was used to check for hydrogen around the weld area. Hydrogen levels in the gap between the inner lid and the canister shell for the first canister were 0.1% hydrogen. During welding, an inline hydrogen monitor (HY-OPTIMA Model 1700) connected to the vent port was used to monitor for hydrogen. This system was sensitive to oxygen and would read high if oxygen was present due to oxygen absorption on the sensor. Purging of the canister with helium was required to remove the oxygen to get an accurate reading on the hydrogen detector. The initial purge level of 1 psig was increased to 2 psig to ensure all the oxygen was being removed from the canister and to further ensure hydrogen levels would remain low. During the welding of the root pass on the inner lid of the first canister, hydrogen levels averaged approximately 0.8% until the weld approached completion, at which time the hydrogen level increased to 1.5%. During the non-destructive examination of the root pass weld

on the inner lid, hydrogen levels coming through the vent port were monitored at 0.91%.

The HY-ALERTA hand held hydrogen monitor, Serial # A000202, used during the initial hydrogen survey of the weld area had a monitoring range of 0.0015% to 100% hydrogen. Calibration had been performed in August 2010. During the loading of the first cask, two inline hydrogen monitors were available. Both were Model HY-OPTIMA 1700 with a range of 0.5% to 100% hydrogen. Hydrogen monitoring was performed by pulling air through the vent port while helium was flowing into the canister through the siphon port. An alarm set point of 2% hydrogen was established to warn of hydrogen gas problems. Hydrogen monitor, Serial # A000153, had been calibrated August 2010. Hydrogen monitor Serial # A000081 had been calibrated July 2010.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (c) Chemistry Procedure 8.5.6 "HY-ALERTA Hydrogen-Specific Detection Instrument," Revision 0, (d) Chemistry Procedure 8.5.7 "HY-OPTIMA 1700 Hydrogen-Specific Detection Instrument," Revision 0, (e) TriVis Procedure 06260-CNS-OPS-01 "Spent Fuel Cask Welding: 61BT HUHOMS Canister," Revision 3

Category: Operations **Topic:** Unintentional Draindown of Transfer Cask

Reference: 10CRFR72.150 Amendment 9

Requirement: In accordance with 10 CFR 72.150, a licensee shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. Contrary to the procedures associated with the transfer cask operations, an unintentional partial draining of the transfer cask neutron shield occurred while containing a loaded canister. The reduction of water shielding resulted in an increase in radiation levels in the area around the cask. The event was the result of failure to follow procedures as required by 10 CFR 72.150.

Observation: On November 3, 2010, Canister CNS61B-005-A was inside the transfer cask in the reactor building railroad airlock area being prepared for movement to the ISFSI pad on the next shift. This was the second canister that had been loaded during this initial loading campaign. The canister contained fuel that ranged in age from 15 to 20.6 years since discharge with an 11.323 kW heat load. The canister had previously been inserted into horizontal storage module (HSM) - 2A on October 29, 2010. During the radiological survey of the inside of the transfer cask after the canister had been inserted into the horizontal storage module, contamination was found that exceeded the Technical Specification 1.2.12 limits. The licensee initiated Condition Report CR-CNS-2010-8093 and retrieved the canister from the horizontal storage module for additional decontamination. The canister was returned to the reactor building refueling floor and cleaned, then lowered to the reactor building railroad airlock area onto the transport trailer ready for final preparations for transport back to the ISFSI. Additional information related to the contamination can be found in this report under the Category: Radiation Protection and Topic: Contamination Survey of Canister.

Prior to transport to the ISFSI pad, workers were directed to ensure all residual water had

been drained from the annulus inside the transfer cask between the canister and the transfer cask wall. The transfer cask was a Transnuclear (TN) Model NUHOMS OS197H. Initial draining of the annulus water was performed while the canister and transfer cask were on the reactor building refueling floor prior to bolting the transfer cask lid in place and lowering the transfer cask/canister to the ground level and placing on the transport trailer. Once the transfer cask was placed horizontally onto the transport trailer, any residual water could be drained from the annulus prior to movement to the ISFSI. Procedure 10.39, Revision 2, effective October 27, 2010 was the procedure in effect when the transfer cask draindown incident occurred. Steps 6.34 through 6.36 provided instructions on draining the annulus while the transfer cask was on the transport trailer and stated "Attach transfer cask/canister annulus drain line and stem fitting to annulus drain fitting. Drain any residual water from the transfer cask/canister annulus to a bucket. Disconnect transfer cask/annulus vent stem fitting and annulus drain line." The transfer cask was configured with three fill and drain ports near the bottom of the transfer cask: (1) the annulus drain line, (2) the neutron shield fill port, and (3) the neutron shield drain port. All three fill and drain ports were located close to one another and used the same size quick connect fittings, which were interchangeable. No labels, tags, warnings, locks or markings were provided to distinguish the ports.

At approximately 5:00 am on November 3, 2010, the transfer cask was moved into the railroad airlock area. Soon afterward, workers proceeded to connect a hose to the annulus drain line going to an empty bucket to allow any excess water to drain from the annulus area. However, the hose leading to the bucket was inadvertently connected to the neutron shield drain port instead of the annulus drain line. When the valve was opened, only air came out. [The description "gulp of air" was used by the worker]. No water entered the bucket. Before the workers left the area, one person observed that there was water that had flowed from the top overflow vent of the neutron shield into the overflow tank. The workers were at the end of their shift. They left the valve open, secured the area and left at about 5:20 am with the remaining workers leaving the area at about 5:30 am. No water was flowing into the bucket.

On the other end of the transfer cask at the top was a neutron shield pressurization and overflow vent fitting. A hose connected this vent to an overflow tank which set on the transport trailer. This was necessary to allow for any expansion of water due to heating from the fuel in the canister during transfer to the ISFSI. When the drain line was opened by the workers, this allowed a siphon to occur which drained water into the overflow tank. The air the workers heard at the bucket was not air coming out, but was air going in to allow for the slow siphoning that was occurring at the other end of the trailer in the overflow tank. At some point after the workers left the area, the siphoning action stopped at the overflow tank and the water began flowing into the bucket that had been left by the workers to collect water from what they thought was the annulus drain line. At approximately 7:00 am, health physics personnel and two other workers returned to the area and found water overflowing from the bucket with 1/2 inch to 3/4 inch of water on the floor under the cask. They closed the neutron shield drain valve and initiated radiological surveys of the cask using portable radiation detectors. Gamma radiation levels were normal at around 15 - 30 mR/hr on contact. Neutron radiation levels around the transfer cask, which were normally around 2 mrem/hr at 30 cm on the side, were readings 15 mrem/hr at 30 cm at ground level. Additional surveys were

initiated of the general area and the upper floors. Radiation levels near the water on the floor did not indicate the water was contaminated. The water was later confirmed as clean.

At 8:13 am, the control room was informed of the partial draindown of the transfer cask neutron shield. The control room ordered an evacuation of the reactor building and the south side of the administrative building. At 8:19 am, the control room entered Abnormal Procedure 5.1RAD based on the reported transfer cask readings of 15 mrem/hr at 30 cm. By 8:25 am, an assessment of the applicable emergency action levels (EAL) had been made and it was determined that no EALs had been exceeded. The control room was provided an update on radiation levels. Normal neutron radiation levels were 10 mrem/hr contact and 2 mrem/hr at 30 cm. Neutron exposure levels were entered into the control room log as 158 mrem/hr contact and 104 mrem/hr at 30 cm based on readings on the upper portion of the transfer cask. Combined with the 15 - 30 mR/hr contact gamma dose rates, the dose rates on the transfer cask were well below the Technical Specification 1.2.11 limit of 500 mrem/hr gamma plus neutron at 3 feet but significantly higher than earlier surveys (before the draindown of the neutron shield).

Follow-up surveys at ground level found gamma and neutron radiation levels normal in the areas around the reactor building and outside. Elevated neutron radiation levels were measured on the upper portions of the transfer cask with slight increases on the floor above and near the hatch. The neutron dose rates on the cask were 205 mR/hr contact and 130 mR/hr at 30 cm as documented on Survey CNS-1011-0008 for the survey taken at 8:22 am. The 130 mrem/hr neutron at 30 cm was the value reported to the NRC in the 24-hour notification report (November 3, 2010). In the sixty-day letter (December 29, 2010), additional dose information was provided. The maximum neutron contact dose was 205 mrem/hr. The maximum gamma contact dose was 30 mR/hr. Converting these doses to three feet gave 104 mrem/hr neutron and 7.5 mrem/hr gamma. The neutron shield was refilled with water by 6:45 pm and ISFSI work was temporarily halted. Approximately 40% (220.8 gallons) of the neutron shield volume had drained. Estimated dose to the 20 workers involved with the incident was 47.4 mrem.

The unintentional draining of the transfer cask was reportable as a non-emergency 24-hour notification under 10 CFR 72.75(d). This regulation required that each licensee shall notify the NRC within 24 hours after discovery of an event in which important-to-safety equipment is disabled or fails to function as designed when (i) the equipment is required by regulation, license condition, or certificate of compliance to be available and operate to prevent releases that could exceed regulatory limits, to prevent exposure to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident, and (ii) no redundant equipment was available and operable to perform the required safety function. The NUHOMS Updated Final Safety Analysis Report (UFSAR), Section 3.4.4.1 "Transfer Cask and Yoke," and Section 11.2 "Important-to-Safety and Safety Related NUHOMS System Components," identified the transfer cask as important-to-safety. UFSAR Sections 4.7.3.2 "Transfer Cask," Section 8.2.5.3 "Accident Dose Calculations for Loss of Neutron Shield," and Section K.11.2.5.3 "Accident Dose Calculations for Loss of Neutron Shield," provided an analysis of the radiological consequences to workers from a loss of the neutron shield. For a canister containing the maximum allowed fuel at 18.3 kW, Section 8.2.5.3.2 calculated that the

cask surface dose rate could increase from 552 mrem/hr to 2128 mrem/hr when the neutron shield is lost. The licensee reported in their root cause analysis that the dose rates with all the neutron shield water lost could have reached a contact dose of 600 mrem/hr neutron and 400 mrem/hr gamma for the 11.4 kW fuel in the canister. The licensee submitted to the NRC an Event Notification Report (Event Number 46391) on November 3, 2010 at 16:36 EST. This met the 24 hour notification requirement. In addition, a 60 day follow-up report was required by 10 CFR 72.75(g). This report was submitted to the NRC on December 29, 2010 within the 60 days.

Condition Report CR-CNS-2010-08192 was issued as a Category A condition report on November 3, 2010. As a Category A condition report, a root cause analysis was required. The root cause analysis was completed and issued November 30, 2010 and provided a thorough evaluation of the incident. The root cause for the event was determined to be human factors deficiencies that were inadvertently designed into the equipment. The root cause analysis identified a number of corrective actions. These included labeling the three drain ports, securing the neutron shield ports such that they could not be operated after the neutron shield jacket was filled with water, providing additional training to the workers, and revising several procedures adding additional checks and clarifications to prevent recurrence and requiring continuous monitoring during annulus draining activities. Revision 7 to Procedure 10.39, Attachment 6 "Transfer Cask Lower Neutron Shield Jacket Fitting" included pictures of the drain and fill fittings and clearly marked which port was associated with the transfer cask neutron shield and which was the annulus drain port. A new Attachment 7 "Transfer Cask Neutron Shield Fitting with RP Control Tag/Fitting ID and Installed Shrink Wrap," was also added to provide additional clarification. The licensee reviewed the INPO Operating Experience database and found no similar events. The licensee noted that over 500 NUHOMS casks have been loaded to-date.

Several additional condition reports were issued including CR-CNS-2010-08210 concerning entry into Abnormal Procedure 5.1RAD and the implementation of Nuclear Performance Procedure 10.51 related to the abnormal condition resulting from the partial draindown. Condition Report CR-CNS-2010-8219 addressed the need to revise Procedure 10.51 to incorporate actions for refilling the neutron shield. Procedure 10.51, Section 4.2.7 "Loss of the Neutron Shield," directed that a radiological survey be performed of the transfer cask and a boundary established to limit personnel access if the neutron shield was lost. An action plan to re-establish the neutron shield would then be developed. Condition Report CR-CNS-2010-08219 directed that the action plan be developed and incorporated into the procedural steps. Additional actions were added to Section 4.2.7 with specific instruction on how to refill the neutron shield. Transnuclear Inc. was notified of the incident and entered the condition into their corrective action program.

The unintentional draindown of the loaded transfer cask was the result of workers not following procedures. 10 CFR 72.150 "Instructions, Procedures and Drawings" states "The licensee, applicant for a license, certificate holder and applicant for a Certificate of Compliance shall prescribe activities affecting quality by documented instructions, procedures and drawings of a type appropriate to the circumstances and shall require that these instructions, procedures and drawings be followed." In addition, 10 CFR

72.212(b)(9) states "Conduct activities related to the storage of spent fuel under this general license only in accordance with written procedures." Contrary to this, on November 3, 2010, workers did not follow Procedure 10.39, Revision 2, Steps 6.34 through 6.36 and connected a drain line to the wrong drain fitting. As a consequences of this action, approximately 220 gallons (40%) of the water from the transfer cask neutron shield was unintentionally drained, resulting in increased radiation levels around the cask. Because this violation was self identified and not repetitive or willful, the issue was entered into your corrective action program, and compliance was restored, this violation is being treated as a Severity Level IV non-cited violation (NCV) consistent with the NRC Enforcement Manual, Section 2.3.2.

Documents Reviewed: (a) Nuclear Performance Procedure 10.39 "Dry Shielded Canister Transport from Reactor Building to ISFSI," Revision 2, Revision 7, and Revision 8, (b) Nuclear Performance Procedure 10.51 "ISFSI/DFS Abnormal Operations," Revision 0 and Revision 1, (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10, (d) Condition Report CR-CNS-2010-8093 "Unexpected Contamination Levels Found Inside Transfer Cask," initiated October 30, 2010, (e) Condition Report CR-CNS-2010-08192 "Transfer Cask Neutron Shield Drain Valve Left Open Causing Partial Loss of Neutron Shield," initiated November 3, 2010, (f) Condition Report CR-CNS-2010-8210 "Entry into Abnormal Procedure 5.1RAD," initiated November 3, 2010, (g) Condition Report CR-CNS-2010-08219 "NPP 10.51 Abnormal Procedure Guidance Inadequate," initiated December 4, 2010, (h) Radiological Survey CNS-1011-0008 of the transfer cask after the partial drain down of the neutron shield, dated November 3, 2011 at 8:22 am, (i) Control Room Log, dayshift for November 3, 2010 from 8:13 am to 10:35 am, (j) Event Notification Report (EN) #46391 "Fuel Storage Transfer Cask Neutron Shield Partial Drain-Down," dated November 3, 2010 at 16:36 ET, (k) Cooper Nuclear Station Root Cause Investigation Report "ISFSI Transfer Cask Neutron Shield Drain Valve Left Open," dated November 30, 2010, (l) Letter (NLS2010111) from Demetrius L Willis, Nebraska Public Power District to NRC Document Control Desk entitled "Independent Spent Fuel Storage Installation Sixty-Day Follow-up Report, Cooper Nuclear Station, Docket 50-298, DPR-46, Cooper Nuclear Station ISFSI, Docket No. 72-066, " dated December 29, 2010 [NRC ADAMS Accession No. ML110050081]

Category: Pre-Operational Test **Topic:** Pre-Operational Testing Requirements

Reference: CoC 1004, Tech Spec 1.1.6; UFSAR 1004, Sect 9.2 Amendment 9/Rev. 10

Requirement: A dry run of the canister loading, transfer cask handling, and canister insertion into the horizontal storage module (HSM) shall be held. The dry run shall include: 1) functional testing of the transfer cask and lifting yoke; 2) loading the canister into the transfer cask and installing the annulus seal; 3) transporting the transfer cask to the ISFSI with the transfer trailer and aligning it with the HSM; 4) inserting a weighted canister into the HSM and retrieving it; 5) loading a mock-up fuel assembly into the canister; 6) sealing, vacuum drying and helium backfilling of a (mock) canister; 7) opening a (mock) canister; and 8) returning the canister and transfer cask to the spent fuel pool.

Observation: The licensee completed all the required pre-operational testing requirements during two NRC observed dry runs performed during the weeks of February 23, 2009 and September

27, 2010. Technical Specification 1.1.6 listed eight specific demonstrations that were required. Demonstration #6 (canister sealing, vacuum drying, and cover gas backfilling operations) and Demonstration #7 (opening a canister) were demonstrated on February 23, 2009 and documented in Inspection Report 72-66/2009-01, dated August 28, 2009. The remaining demonstrations, #1 through #5 and #8, were performed at the Cooper nuclear facility during the week of September 27, 2010. Demonstration #1 required a functional test of the transfer cask with the lift yoke to ensure the transfer cask could be moved between the spent fuel pool, the decontamination area, and the transport trailer. This was successfully demonstrated with a transfer cask containing a canister fully loaded with dummy fuel elements on September 27 and October 1, 2010. Demonstration #2 required loading a canister into the transfer cask, verifying the fit of the canister, and verifying the fit of the annulus seal. This was demonstrated on September 30, 2010. Demonstration #3 required transporting the transfer cask on the transport trailer from the reactor building to the ISFSI and aligning the transfer cask with the horizontal storage module opening. This was demonstrated on September 27, 2010 when the licensee down-ended the weighted transfer cask onto the transport trailer, transported the transfer cask from the reactor building to the ISFSI, and aligned the transfer cask to the horizontal storage module. Demonstration #4 required testing the transport trailer alignment and docking equipment, inserting a canister into a horizontal storage module, and then retrieving it. This was demonstrated on September 27, 2010 and included the insertion of a weighted cask into the horizontal storage module, disconnecting, then offsetting the transport trailer from the horizontal storage module, re-aligning, and then retrieving the canister from the horizontal storage module back into the transfer cask. Demonstration #5 required loading a dummy fuel assembly into the canister. This demonstration was performed on October 1, 2010. The licensee placed a mock-up fuel assembly into the four corners of the canister to show that the fuel handling crane could reach all locations. Additionally, the licensee un-grappled the dummy fuel element, offset the crane, and then re-grappled to demonstrate the ability to unload a canister, if ever needed. Demonstration #8 required placing the canister and transfer cask into the spent fuel pool. The licensee performed this on October 1, 2010, by placing the transfer cask loaded with a canister into the spent fuel pool prior to the dummy fuel element loading demonstration.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (c) NRC Inspection Report 07200066/2009001 [NRC ADAMS Accession No. ML092430509], dated August 28, 2009

Category: Quality Assurance **Topic:** Approved QA Program

Reference: 10 CFR 72.140(d) Published 2010

Requirement: A QA program previously approved by the Commission as satisfying the requirements of Appendix B of Part 50 will be accepted as satisfying the requirements of Part 72. In filing the description of the QA program required by Part 72.140(c), each licensee shall notify the NRC of it's intent to apply it's previously approved QA program to ISFSI activities. The notification shall identify the previously approved QA program by date of

submittal, docket number and date of Commission approval.

Observation: Nebraska Public Power District notified the NRC on November 1, 2007 of their intent to apply the Part 50 QA program to the ISFSI activities in a letter to the NRC Document Control Desk. The original QA program had been accepted by the NRC as Amendment 39 to the Cooper Nuclear Station Final Safety Analysis Report, Docket No. 50-298.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Letter (NLS2007076) from David W. Van Der Kamp, Nebraska Public Power District to the Director, NRC Spent Fuel Project Office entitled "Notification of Intent to Apply Previously Approved 10 CFR 50 Quality Assurance Program to Independent Spent Fuel Storage Installation Activities Cooper Nuclear Station, Docket No. 50-298, DPR-46," dated November 1, 2007 [NRC ADAMS Accession No. ML073120011]

Category: Quality Assurance **Topic:** Control of Measuring and Test Equipment

Reference: 10 CFR 72.164 Published 2010

Requirement: The licensee shall establish measures to ensure that tools, gauges, instruments and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specific periods to maintain accuracy within necessary limits.

Observation: The licensee had established requirements for calibration of equipment used on the ISFSI project in Administrative Procedures 0.37 and 0.38. During the loading of the first canister, calibration of equipment was evident by calibration stickers on the equipment. All equipment verified by the NRC inspectors were found to be within the specified calibration dates.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Administrative Procedure 0.37 "Measuring and Test Equipment (M&TE) Calibration Program Guidelines," Revision 24, (c) Administrative Procedure 0.38 "Processing Instrument Calibration Program," Revision 5

Category: Quality Assurance **Topic:** Corrective Actions

Reference: 10 CFR 72.172 Published 2010

Requirement: The licensee shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action taken to preclude repetition. This must be documented and reported to appropriate levels of management.

Observation: A broad range of condition reports were reviewed. The licensee provided a list to the NRC inspectors of all the condition reports that had been issued related to the dry cask storage program and the reactor building crane. From the list, selected condition reports were identified for review, including any root cause analysis that had been performed. Those that related to specific topics covered in this inspection report are further discussed under the appropriate heading for the issue as documented in these Inspector Notes. The reviews found that the licensee was readily identifying issues early and documenting them in the corrective action program for resolution and trending. The

employees were encouraged by management to use the corrective action system and to document concerns so that they received the proper level of attention, including that of management. The review by the NRC inspectors resulted in several meetings to discuss actions that had been taken to close the condition report. No concerns were identified related to the closure of the condition reports reviewed.

The following provided a summary of the condition reports reviewed. The crane had a number of issues in preparation for the loading of the first cask. Condition Report CR-CNS-2006-04655 documented that the reactor building crane, while raising the yoke back to it's top elevation, only traveled about 11 feet and stopped. Condition Report CR-CNS-2006-06648 documented the discovery of a crack on the reactor building crane east bridge railing. The crack was on an end rail splice weld. The weld was part of the original plant construction. Similar cracks had been previously found and repaired at other rail locations. Condition Report CR-CNS-2007-03844 documented that the reactor building crane trolley would not move. Condition Report CR-CNS-2008-00154 documented that while moving cribbing, the reactor building crane bridge stopped and would not move north or south. Condition Report CR-CNS-2008-04810 documented a concern about the seismic calculations for the crane related to the limitation of 70 tons placed on the crane rated load. Condition Report CR-CNS-2008-07207 documented that during acceptance testing, the crane could not lift the 100% load without drawing more than the full load amperage of 110% to 115% of the nameplate full load current. Condition Report CR-CNS-2008-07968 documented that a Whiting Corp. design analysis and study of the crane had identified two overstress conditions. One was on the girder end connection of the low head room bridge and one on a single fastener in a main hoist gear case application. Condition Report CR-CNS-2009-02495 discussed a concern related to the crane analysis for the design basis earthquake and design basis tornado. Condition Report CR-CNS-2010-06665 discussed an observation during the lowering of the crane hook that the load block did not lower in a smooth fashion, indicating a lubrication issue. Condition Report CR-CNS-2010-06702 documented questionable readings on the load cell while holding the yoke. The reading varied from 3,000 to 12,000 pounds and should have read approximately 9,600 pounds. Condition Report CR-CNS-2010-06815 documented an inconsistency between the requirement in the NUHOMS Updated Final Safety Analysis Report (UFSAR), page 4.2-11 and the American National Standards Institutes (ANSI) N14.6 guidance for special lifting devices related to the 300% load test and the non-destructive testing requirements for the lift yoke. Condition Report CR-CNS-2010-07669 documented several inconsistencies between the wire rope inspection procedures with the recommendations of the American Society of Mechanical Engineers (ASME) B30.2 "Overhead Gantry Cranes." Condition Report CR-CNS-2010-09447 documented the discovery of burnt wiring on the crane's dynamic braking resistor. The wires had come loose from their holding strap and made contact with a resistor that typically gets very hot.

Condition Reports CR-CNS-2010-9089 and CR-CNS-2010-09222 documented that two horizontal storage modules exceeded the technical specification temperature limits following insertion of the canisters. The limits were slightly exceeded and could be attributed to cold ambient temperatures and sudden temperature swings. Condition Report CR-CNS-2010-06737 related to improving the training program. Condition Reports CR-CNS-2010-06809 and CR-CNS-2010-06810 discussed the need to complete

documents prior to the loading campaign. Condition Report CR-CNS-2010-09609 documented that during the lowering of Cask #7 into the spent fuel pool, water entered the ductwork above the pool skimmers and overflowed onto the floor at elevation 976 feet. Previous casks had been lowered into the spent fuel pool over a 20-25 minute period, which allowed the water to stabilize. Cask #7 was lowered into the pool in approximately 12 minutes, causing a high level alarm on the water level monitor for the spent fuel pool. Condition Report CR-CNS-2009-05966 and CR-CNS-2010-05987 dealt with calibrated instrument issues.

Condition Reports CR-CNS-2009-03321, CR-CNS-2009-03325, and CR-CNS-2009-03328 documented issues with the acceptance of certain linear indications shown on the radiographs of the fabrication welds on Canisters #4, #7, and #8. The indications had been originally determined to be non-relevant and classified as "ghost images" by the vendor. However, acceptance review by the Cooper Nuclear Station Maintenance Welding Coordinator determined that the indications did not comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1998-99 edition. The code stated that linear indications exceeding 1/4 inch for this thickness range of metal was unacceptable. The welds were repaired for Canisters #7 and #8. Canister #4 was rejected and replaced. Condition Report CR-CNS-2010-05053 was initiated to document the replacement of Canister #4 with a replacement canister.

Condition Report CR-CNS-2009-00729 discussed the use of grout for cosmetic repairs of the horizontal storage modules and stated that documentation had not been developed to verify that the density of the grout was acceptable for repairs. The density of the horizontal storage module was important to the shielding analysis in the final safety analysis report. A specific grout for use on the modules had been identified by the vendor, but a comparison of the grout's density to that of the original concrete in the horizontal storage module had not been documented.

Condition Report CR-CNS-2010-07719 documented scratches found on the outer surface of Canister CNS61B-002-A. This canister had been used during the dry run demonstration for the insertion into the horizontal storage module and had been scratched by the rails in the horizontal storage module. The minimum design thickness requirement for the canister shell was 0.490 inches per TN Drawing 10961-30-13. Several scratches were reported with the worst scratch at a depth of 0.074 inches. The fabrication records for Canister CNS61B-002-A documented a minimum shell wall thickness for the canister as 0.508 inches. Subtracting the 0.074 inches reduced the shell thickness to 0.434 inches in the scratched area which was below the design thickness of 0.490 inches. The vendor, Transnuclear Inc. was contacted and issued Nonconformance Report 2010-186. This issue had been addressed in calculations previously performed by Transnuclear, Inc. which calculated the maximum allowable scratch depth for a 61BT canister. The calculations were documented in TN Calculation 1093-102, Revision 0. The calculation had assumed a minimum shell wall thickness of 0.40 inches due to a scratch. The analysis found that the stresses on the canister were still within the permissible stresses allowed by the the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NB (1998 edition including the 1999 addenda). As such, the canister could be dispositioned as "use-as-is." Engineering Evaluation 10-066 was performed by Cooper Nuclear Station which

determined the scratches to be surface flaws with no impact on the critical design function of the cask. Cooper accepted the Transnuclear Inc. evaluation to use-as-is. A 72.48 screening was completed of the engineering analysis and Transnuclear's conclusions.

On October 1, 2009, Transnuclear Inc. notified the NRC of a potential Part 21 violation related to a number of casks previously delivered to reactor sites that had various weld studs, washers, screws, port plugs and other small components that were found to have falsified documentation of their material content needed to demonstrate compliance with required design specifications. Seven of these canisters had been delivered to the Cooper site, though none had been loaded with spent fuel. The notification was documented in NRC Event # 45398. A follow-up report was issued by Transnuclear Inc. on October 30, 2009. Cooper Nuclear Station issued Condition Report CR-CNS-2009-07489 to document that Hwa Shin Bolt Industrial Company of South Korea may have provided unsubstantiated certified material test reports (CMTRs) for certain small parts utilized in the fabrication process for the canisters used at Cooper. None of the affected casks had been loaded at Cooper at the time of notification. Transnuclear Inc. initiated Corrective Action Report (CAR) No. 2009-086 and performed an engineering evaluation and significant safety hazards determination of the affects of the parts being used in the canisters and transfer cask. The analysis determined that the parts that had been supplied were similar in strength and corrosion resistance as the ones originally specified and that failure of the parts would not affect public health and safety. Each part was analyzed in accordance with it's function. Transnuclear Inc. concluded that some of the parts were only necessary during fabrication and were not important after the basket was assembled, other parts would not be affected by the change in material, and the test port plugs were subject to visual and liquid penetrant examination after welding that would confirm their acceptance. Transnuclear Inc. committed to replace all suspect parts for all canisters and transfer casks if replacement would not cause damage to the canister. For any canisters that had already been loaded, replacement was not possible and Transnuclear, Inc. would provided documented justification for not removing the items.

Condition Report CR-CNS-2010-08093 documented the discovery of unexpected contamination levels inside the transfer cask after the second canister had been inserted into the horizontal storage module. A discussion of this is provided in these Inspector Notes under the Category: Radiation Protection and the Topic: Contamination Survey of Canister. Condition Reports CR-CNS-2010-08192, CR-CNS-2010-08210, and CR-CNS-2010-08219 documented the draindown of the loaded transfer cask while inside the reactor building railroad airlock area. This issue is discussed in these Inspector Notes under the Category: Operations and the Topic: Unintentional Draindown of Transfer Cask.

In addition to the corrective action reports discussed above, Cooper stayed current with issues concerning the Transnuclear casks and participated in discussions with other users. When issues were identified that could affect the Cooper program, they were evaluated and changes made to their program. One of these issues was the potential over-pressurization of a canister by Southern California Edison Company [NRC ADAMS Accession No. ML111430612]. The subsequent investigation determined that the canister had not over-pressurized, however, Cooper reviewed the incident and wrote a

"white paper" that documented their review of whether a similar event could occur during the loading campaign at Cooper. Warnings were added to the appropriate procedures.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Condition Report CR-CNS-2006-04655 "Crane Unexpectedly Stopped During Lift of Yoke," June 28, 2006, (c) Condition Report CR-CNS-2006-06648 "Crack Found on Reactor Building Crane Rail," initiated September 15, 2006, (d) Condition Report CR-CNS-2007-03844 "Reactor Building Trolley Will Not Move," initiated May 30, 2007, (e) Condition Report CR-CNS-2008-00154 "Crane Will Not Move North or South," initiated January 8, 2008, (f) Condition Report CR-CNS-2008-04810 "CNS Reactor Building Crane Seismic Upgrade for ISFSI Load," initiated June 19, 2008, (g) Condition Report CR-CNS-2008-07207 "During 100% Load Test, Crane Drawing Excess Amperage," initiated September 24, 2008, (h) Condition Report CR-CNS-2008-07968 "Some Structures of the Reactor Building Crane May Not Be Adequately Designed to Mitigate Damage During Design Basis Events," dated October 29, 2008, (i) Condition Report CR-CNS-2009-00729 "Evaluation of Grout Used for Repairs on HSM," initiated January 29, 2009, (j) Condition Report CR-CNS-2009-02495 "Reactor Building Crane Seismic Analysis," initiated March 26, 2009, (k) Condition Report CR-CNS-2009-03321 "Radiographs of Welds on Canister #7," initiated April 27, 2009, (l) Condition Report CR-CNS-2009-03325 "Radiographs of Welds on Canister #8," initiated April 27, 2009, (m) Condition Report CR-CNS-2009-03328 "Radiographs of Welds on Canister #4," initiated April 27, 2009, (n) Condition Report CR-CNS-2009-05966 "Calibrated Instrument Provided by Vendor Not on Approved List," initiated August 7, 2009, (o) Condition Report CR-CNS-2009-07489 "Transnuclear Part 21 EN45398 - Fasteners on Spent Fuel Storage Devices Did Not Meet Standards," dated October 2, 2009, (p) Condition Report CR-CNS-2010-05053 "Replacement Canister #4," initiated January 7, 2011, (q) Condition Report CR-CNS-2010-05987 "Calibration of Instrument Used for DSC Alignment," initiated August 19, 2010, (r) Condition Report CR-CNS-2010-06665 "Load Block Did Not Lower Smoothly," initiated September 13, 2010, (s) Condition Report CR-CNS-2010-06702 "Load Cell Reading May Not Be Correct," initiated September 14, 2010, (t) Condition Report CR-CNS-2010-06737 "Training Requirements for Refuel Bridge Operator," initiated September 15, 2010, (u) Condition Report CR-CNS-2010-06809 "Document EE 09-011 Needs to Be Completed," initiated September 17, 2010, (v) Condition Report CR-CNS-2010-06810 "Fire Hazards Analysis Needs to Be Completed," initiated September 17, 2010, (w) Condition Report CR-CNS-2010-06815 "TN UFSAR Inconsistency with ANSI N14.6," initiated September 17, 2010, (x) Condition Report CR-CNS-2010-07669 "Procedures Concerning Wire Rope Inspection are not Consistent with ASME B30.2," initiated October 15, 2010, (y) Condition Report CR-CNS-2010-07719 "Scratches on Canister #2," initiated October 18, 2010, (z) Condition Report CR-CNS-2010-08093 "Unexpected Contamination Levels Found Inside Transfer Cask," initiated October 30, 2010, (aa) Condition Report CR-CNS-2010-08192 "Transfer Cask Neutron Shield Drain Valve Left Open Causing Partial Loss of Neutron Shield," initiated November 3, 2010, (bb) Condition Report CR-CNS-2010-08210 "Entry into Abnormal Procedure 5.1RAD," initiated November 3, 2010, (cc) Condition Report CR-CNS-2010-08219 "NPP 10.51 Abnormal Procedure Guidance Inadequate," initiated December 4, 2010, (dd) Condition Report CR-CNS-2010-09089 "HSM 1A Delta Temperature Exceeded," initiated December 7, 2010, (ee) Condition Report CR-CNS-2010-09222 "HSM 1B Delta

Temperature Exceeded," dated December 11, 2010, (ff) Condition Report CR-CNS-2010-09447 "Burnt Wiring on Brake Resistors on Crane," initiated December 21, 2010, (gg) Condition Report CR-CNS-2010-09609 "Spent Fuel Pool High Level Alarm," initiated December 27, 2010, (hh) NRC Event Report #41318 "Overstress Condition on Single Failure Proof Crane Trolleys," reported by Whiting Corp. on January 7, 2005 (ii) NRC Event Report #42461 "Crane Overstress Conditions," reported by Whiting Corp. on March 31, 2010, (jj) NRC Event Report #45398 "Fasteners on Spent Fuel Storage Devices Did Not Meet Standards" reported by Transnuclear Inc. as a Part 21 Notification on October 1, 2009, (kk) Transnuclear, Inc. Part 21 Thirty Day Report E-28731 "Report to the USNRC Unsubstantiated Certified Material Test Reports Hwa Shin Bolt Industrial Co., Ltd.," dated October 30, 2009 (NRC ADAMS Accession No. ML093060395), (ll) Transnuclear, Inc. Nonconformance Report (NCR) No. 2010-186 "Scratches on Canister #2 at Cooper Nuclear Station," Revision 1, (mm) CNS Engineering Evaluation #10-066 "Dry Shielded Canister #02," dated December 15, 2010, (nn) CNS Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 11

Category: Quality Assurance **Topic:** Important-to-Safety Items

Reference: UFSAR 1004, Section 11 & Appendix K.2.3 Revision 10

Requirement: The Quality Assurance Program is to be applied to "important-to-safety" and "safety related" activities associated with the standardized NUHOMS system.

Observation: Procedure 0.19.1 provided the criteria and methodology for determining the safety and quality classification of systems, components, and structures for the ISFSI project. Important-to-safety (ITS) was defined in Procedure 1.4QA, Section 2.5 "Definitions" and referenced Regulatory Guide 7.10. Regulatory Guide 7.10 provided an acceptable approach to classifying items in a graded approach based on their relative safety significance. Cooper had adopted this concept in Procedure 0.19.1 and had described three categories consistent with Regulatory Guide 7.10. Procedure 0.19.1, Section 2.5 defined ITS-A as critical to safe operations, ITS-B as having a major impact on safety, and ITS-C as having a minor impact on safety. Remaining structures, systems, and components and consumable items whose failure or malfunction would not impact safety were designated as not important-to-safety (NITS).

Procedure 0.19.1 Section 4.0 “Instructions,” stated that the dry fuel storage system components shall use the quality classifications determined in the latest revision of Report 11301-0100 "NUHOMS 61BT Critical Attributes Report for the HSM-202 and 61BT Canister." Structures, systems, and components for the dry fuel storage system used to support Transnuclear equipment at Cooper were required to be analyzed per Procedure 0.19 in order to determine whether their function or physical characteristics were important-to-safety.” Procedure 0.19, Attachment 4 "Classification Evaluation" provided questions to ask during the evaluation to determine which safety classification should be assigned. The NUHOMS Updated Final Safety Analysis Report (UFSAR), Table 3.4-1 "NUHOMS Major Components and Safety Classification" provided a listing of the various components associated with the canister, transfer cask, and storage cask and designated each component as to their safety classification as important-to-safety or not important-to-safety. UFSAR Section 11.2 “Important-to-Safety and Safety Related NUHOMS System Components” described the same systematic approach of rating

components (into A, B, C, or NITS) based on Regulatory Guide 7.10. The questions for determining an item's category using Cooper's Procedure 0.19 met the guidance of UFSAR Section 11.2. Appendix K "NUHOMS 61BT System" of the UFSAR provided specific information for the 61BT canister. Section K.2.3 stated that components of the 61BT canister that were important-to-safety and not important-to-safety were listed in Table K.2-8 "Classification of NUHOMS DSC Components." Table K.2-8 provided a list of the various components associated with the canister that were important-to-safety or not important-to-safety.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Administrative Procedure 0.19 "Equipment and Record Functional Location File Program," Revision 25, (c) Site Services Procedure 1.4QA "Quality Assurance/Safety Classification Requirements," Revision 5, (d) Administrative Procedure 0.19.1 "Quality Assurance Program Applicability to Dry Fuel Storage," Revision 1

Category: Quality Assurance **Topic:** QA Audits

Reference: 10 CFR 72.176 Published 2010

Requirement: The licensee shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the program.

Observation: Cooper had incorporated the dry cask storage activities into their quality assurance program. The 2010 audit schedule listed an audit of the ISFSI activities conducted on June 7, 2010. The Audit Element/Attribute Matrix listed the areas included in the audit as the ISFSI program and licensing requirements, ISFSI design control, ISFSI operations, and ISFSI maintenance. The resulting QA Audit Report 10-03 determined that the Cooper Nuclear Station Part 50 audit program had been successfully implemented for the dry cask storage activities. The ISFSI activities were designated to be audited every 2 years. Audit 10-03 was the first audit of the ISFSI activities. Many corrective actions were identified in the report in which the licensee issued numerous condition reports to track and adequately resolve the concerns. In addition to the QA audit, multiple surveillance activities had been completed on the dry cask storage activities. The resulting issues identified during the surveillance activities had been placed into the licensee's corrective action program for resolution.

Cooper Nuclear Station personnel had also performed multiple surveillances of Transnuclear fabrication activities at Hitachi-Zosen (TN's Fabricator) for the NUHOMS 61BT canisters. NRC inspectors reviewed the surveillance plans, which covered areas such as: final document package review, oversight of ISFSI canister fabrication and welding activities, and material procurement controls. Findings that were identified were documented as being discussed with the Transnuclear QA representative and Hitachi Zosen QA representative for resolution.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) NPPD QAD20100024 "Year 2010 Audit Schedule," Revision 3, (c) Cooper Nuclear Station (CNS) Quality Assurance Master Audit Plan (MAP) - Audit Element/Attribute Matrix, dated August 11, 2010, (d) QA Audit 10-03, "Radiological Material Processing &

Shipping, Radiological Protection, and Independent Spent Fuel Storage Installation," (QAD20100026), dated August 9, 2010, (e) S08-05 "Dry Fuel Storage/Independent Spent Fuel Storage Installation (ISFSI)," (QAD 2008036), dated May 28, 2008, (f) S08-06 "Dry Fuel Storage /Independent Spent Fuel Storage Installation (ISFSI)," (QAD 2008021), dated March 13, 2008, (g) S07-19 "Dry Fuel Storage /Independent Spent Fuel Storage Installation (ISFSI)," (QAD 20070068), dated October 2, 2007, (h) NPPD Supplier Surveillance Report No. SS08-020, Surveillance Period: November 4-11, 2008, (i) NPPD Supplier Surveillance Report No. SS10-008, Surveillance Period: June 29- July 2, 2010

Category: Radiation Protection **Topic:** ALARA

Reference: 10 CFR 72.104(b) Published 2010

Requirement: Operational restrictions must be established to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI operations

Observation: The licensee's ALARA plan for cask loading operations and the radiological protection job plan for ISFSI discussed high dose rate issues associated with dry cask loading activities. Exposure and person-rem estimates for various evolutions were provided based on calculations and experience at other sites during similar loading activities. As presented in the Calculation of Person-Rem Worksheets for Radiation Work Permits 2010-37 and 2010-114, total person-rem estimated for the entire loading campaign for all eight casks was 7.4 person-rem. Estimates for individual cask loadings were also provided. For the first cask, the estimated dose was 0.8 person-rem gamma and 0.6 person-rem neutron for a total of 1.4 person-rem. The actual dose received by the workers for the first cask loading was 0.7 person-rem, of which, nearly all the dose was from gamma exposure.

ALARA lessons learned from ISFSI campaigns at other sites and lessons learned from the licensee's site specific dry runs were used to improve procedures and to focus radiation protection oversight during certain activities that had the potential for unnecessary personnel exposures. Cameras and telemetry were utilized to aid radiation protection personnel in job coverage. The radiological data management system maintained a current record of personnel dose as a function of several parameters. This data was easily accessible and was used to perform periodic evaluations and track personnel exposures.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, b) ALARA Plan for Dry Cask Loading, Revision 0, (c) Radiological Protection Job Plan for ISFSI, Revision 0, (d) Radiation Protection Procedure 9.ALARA.1 "Personnel Dosimetry and Occupational Radiation Exposure Program," Revision 38, (e) Calculation of Person-Rem Worksheets for Radiation Work Permit 2010-37 and 2010-114, dated May 22, 2010

Category: Radiation Protection **Topic:** Berms and Shield Walls

Reference: CoC 1004, Tech Spec 1.1.9 Amendment 9

Requirement: When supplemental shielding and engineered features, such as the berm around the ISFSI pad and the shield walls, are used to ensure that the requirements of 10 CFR

72.104(a) are met, such features are to be considered important-to-safety and must be evaluated under 72.212(b).

Observation: No berms or shield walls were used to reduce exposures around the ISFSI. The ISFSI was located within the Cooper Nuclear Station protected area. Other than the concrete modules themselves and the storage canister, there were no other engineering features that were used for compliance with the requirements of 10 CFR 72.104 (a).

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0

Category: Radiation Protection **Topic:** Contamination Survey of Canister

Reference: CoC 1004, Tech Spec 1.2.12; UFSAR Sect 3.3.7.1.3 Amendment 9/Rev. 10

Requirement: Following placement of each loaded transfer cask into the cask decontamination area, fuel pool water above the shield plug shall be removed and the top region of the of the canister and cask shall be decontaminated. A contamination survey of the upper one foot of the canister shall be taken. The canister smearable surface contamination levels on the outer surface of the canister shall be less than 2,200 dpm/100 square cm from beta and gamma emitting sources and less than 220 dpm/100 square cm from alpha emitting sources.

Observation: The contamination limits specified in Technical Specification 1.2.12 were incorporated into Procedures 10.38 and 10.40. Procedure 10.38, Step 4.62 directed the radiation protection staff to dry and decontaminate the cask after it had been moved from the spent fuel pool to the cask washdown area. Steps 4.85 through 4.93 discussed the removal of the annulus seal and the survey of the top one foot of the canister. The purpose of this survey was to verify the annulus seal had not leaked and allowed water from the spent fuel pool to contaminate the canister. The removable contamination limits specified in Technical Specification 1.2.12 were incorporated into Step 4.92. A contamination survey was performed of the first cask in accordance with Procedure 10.38. Contamination levels on the top one foot of the canister were documented as zero disintegrations per minute (dpm)/100 square centimeters (cm) alpha and 223 dpm/100 square cm beta/gamma which met the 220 dpm/100 square cm alpha limit and the 2,200 dpm/100 square cm beta/gamma limit.

The licensee had incorporated several specific actions into the procedures to clean the cask so that workers would not have to wear protective clothing while working around the cask. As the cask containing the loaded canister was being removed from the spent fuel pool, the top of the cask and canister were pressure washed (Step 4.30). At that time, the annulus seal was still in place. The cask was placed in the cask washdown area and dried and decontaminated (Step 4.82). The annulus seal was removed (Step 4.85), the annulus water was drained to one foot or lower (Step 4.89.2), and the survey to comply with Technical Specification 1.2.12 was performed (Steps 4.91 and 4.92). Upon completion of the radiological contamination survey, the radiation protection representative and the cask loading supervisor signed-off that the technical specification limit was met. Between Step 4.89.2 to drain the water in the annulus and Step 4.91 to

perform the survey, there were no instructions to clean the annulus area prior to the contamination survey. As such, the smear survey taken of the top one foot of the annulus area was considered representative of the contamination that may be present on the entire canister outer surface to demonstrate compliance with the Technical Specification 1.2.12 limits. Water was also collected from the annulus area (Step 4.87) and analyzed for contamination concentrations. Only small amounts of Cobolt-60 were detected at 2.65 x 10(-5) microcuries/ml. After the canister had been inserted into the horizontal storage module, a contamination survey of the inside of the transfer cask was performed in accordance with Procedure 10.40, Step 14.24. If contamination was found in the transfer cask, then it was assumed the same amount of contamination was present on the loaded canister that had now been inserted into the horizontal storage module. If contamination levels equaled or exceeded 2,200 dpm/100 square cm beta-gamma or 220 dpm/100 square cm alpha, Step 14.24.1 directed that the radiation protection manager, shift manager and project manager be notified and the transfer cask decontaminated per Step 14.25. A determination was then made concerning actions to take for the canister that had been loaded into the horizontal storage module.

The second canister loaded, Canister CNS61B-005-A, was found to have problems meeting the contamination limits. The initial survey of the upper one foot of the canister was performed on October 24, 2010 and found beta/gamma contamination levels of 558; 234; and 816 dpm/100 square cm with no detectable alpha contamination. These low levels of contamination were below the Technical Specification 1.2.12 limit of 2,200 dpm/100 square cm beta/gamma. The canister was inserted into horizontal storage module HSM-2A on October 29, 2010. During the radiological survey of the inside of the empty transfer cask after the canister had been inserted into the horizontal storage module, contamination was found ranging from 1,007 to 9,707 dpm/100 square cm beta/gamma on large area smears. One hot particle reading 100,000 dpm was found approximately two feet down from the top of the transfer cask. All readings were beta/gamma with no alpha detected. This exceeded the Technical Specification 1.2.12 limit. The licensee initiated Condition Report CR-CNS-2010-08093 and retrieved the canister from the horizontal storage module for further decontamination. Surveys of the retrieved canister found beta/gamma contamination as high as 2,997 dpm/100 square cm on the upper one foot of the canister and as high as 9,690 dpm/100 square cm at four feet from the top. Thirty-nine of the smears taken of the canister down to eleven feet found measurable contamination in excess of 1,000 dpm/100 square cm. The inside of the horizontal storage module, the outlet vents, and the surrounding area was surveyed after canister was removed. All surveys were less than 1,000 dpm/100 square cm. Decontamination was performed by flushing clean water in the annulus between the canister and the transfer cask for several days. Periodic surveys of the canister down to approximately 6 to 7 feet and sampling of the water used for flushing were used to confirm that the contamination had been removed from the canister.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (c) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4, (d) Nuclear Performance Procedure 10.40 "Dry

Shielded Canister Transfer from Transfer Cask to HSM," Revision 4, (e) Condition Report CR-CNS-2010-08093 "Unexpected Contamination Levels Found Inside Transfer Cask," initiated October 30, 2010 and "Apparent Cause Evaluation Report," dated November 23, 2010

Category: Radiation Protection **Topic:** Controlled Area Radiological Doses

Reference: 10 CFR 72.106(a)/(b)/(c) Published 2010

Requirement: For each ISFSI, a controlled area must be established. Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident 5 rem total effective dose equivalent (TEDE) for an accident condition. Minimum distance from the ISFSI to the nearest boundary of the controlled area must be 100 meters. The controlled area may include roads, railroads or waterways as long as arrangements are made to control traffic and protect public.

Observation: The ISFSI was located within the Cooper Nuclear Station protected area inside the existing nuclear power plant owner controlled area. A tour of the Cooper site, to include the ISFSI pad, confirmed that the licensee had established a controlled area with a minimum distance of 800 meters around the ISFSI, well beyond the 100 meter minimum distance. The outermost limit of the controlled area was marked with adequate signs. Appropriate arrangements were made to control traffic and protect public health and safety. The controlled area was traversed by a plant access road and a waterway, the Missouri River. Closure to river traffic, should it be needed, would be performed by the Nebraska Emergency Management Agency (NEMA) under the Cooper Nuclear Station Emergency Plan. NEMA would also restrict road traffic through the controlled area using control points manned by local law enforcement agencies. The dose from the various accidents at the ISFSI were analyzed in the NUHOMS Updated Final Safety Analysis Report (UFSAR), Section K.11.2 "Postulated Accidents." The only accident condition that increased the dose to the owner controlled area was a partial loss of shielding adjacent to a horizontal storage module. Section K.11.2.1.3 "Accident Dose Calculation" provided information on the assumptions for the loss of shielding. The results of the analysis were provided in Table K.11-1 "Comparison of Total Dose Rates for HSM With and Without Adjacent HSM Shielding Effects." The table showed that at 600 meters for a 2 x 10 array of twenty horizontal storage modules placed back-to-back, the dose rate would be 0.0012 mrem/hr under normal conditions and 0.0024 mrem/hr with the loss of the shielding. This was a factor of two difference between the normal dose rate and the accident dose rate. Based on this ratio, Calculation NAI-1313-002 determined that the accident dose at 800 meters from the 2 x 26 array of horizontal storage modules used at Cooper would be twice the normal dose rate of 0.07 mrem/yr, resulting in a dose rate of 0.14 mrem/yr, well below the 5,000 mrem limit. Additional information on the basis for the normal dose rate of 0.07 mrem/yr is provided in these Inspector Notes under the Category: Radiation Protection and the Topic: Evaluation of Effluent/Direct Radiation.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10, (d) Numerical Applications, Inc. Calculation NAI-1313-002 "Cooper Station ISFSI Offsite Accident

Dose and Onsite Occupational Dose for 2 x 26 NUHOMS Facility," Revision 0, (e) Engineering Design Calculation (NEDC) 09-055 "CNS Review of Transnuclear Calculation NAI-1313-002," Revision 0

Category: Radiation Protection **Topic:** Criticality Monitoring

Reference: 10 CFR 72.124(c) Published 2010

Requirement: A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration is not required. The NRC has defined "packaged" to begin when the canister lid is seal welded.

Observation: The licensee provided criticality monitoring of the spent fuel during cask loading, movement, and sealing on the refuel floor using two Eberline Model RMS 3 monitors. Location of the monitors provided the capability to detect a spent fuel criticality during removal of the loaded cask from the spent fuel pool and during operations while the cask was located in the cask washdown area. Placement of the monitors provided for clear audible warning to personnel involved in the dry cask loading evolutions. Installed area monitoring detectors provided backup measurement of radiation levels that would indicate that a criticality had occurred.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Nuclear Performance Procedure 10.36.1 "Fuel Loading/Unloading of a Dry Shielded Canister," Revision 3

Category: Radiation Protection **Topic:** Dose Rate In Empty HSM from Nearby HSMs

Reference: 10 CFR 72.104(b) Published 2010

Requirement: The inside of empty HSM modules have the potential of high dose rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door to the HSM has been removed.

Observation: During the loading of a canister into a horizontal storage module, health physics personnel performed radiological surveys of the work areas. This included the dose rates when the horizontal storage module door was removed. Procedure 10.40, Sections 2.9 and 2.10 required the horizontal storage module door to remain in place except during canister transfer operations, unless a temporary cover was installed. If delays occurred in the transfer after the horizontal storage module door was removed, a temporary cover was required. Step 4.9 of the procedure required a radiological survey prior to any horizontal storage module entry if the adjacent horizontal storage module was loaded with a canister. The Technical Specification 1.2.7 dose rate limits on the horizontal storage module were listed in the procedure, Step 14.18. The radiological protection job plan for the ISFSI, in the section related to the transport of the transfer cask to the ISFSI pad, provided a note that "Dose rates inside a horizontal storage module adjacent to a loaded horizontal storage module will be a radiological concern. Be aware of this when removing a horizontal storage module shield door that is adjacent to a loaded horizontal

storage module." Radiation protection personnel had obtained information from the Monticello nuclear plant, which also used the NUHOMS-61BT cask, concerning expected dose rates adjacent to a loaded horizontal storage module. Highest doses found inside a horizontal storage module adjacent to a loaded one was 100 to 300 mrem/hr. This was at the ventilation space where the dose would be the highest. General area inside the adjacent horizontal storage module was less than 100 mrem/hr. The adjacent empty horizontal storage module was posted as a high radiation area.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Nuclear Performance Procedure 10.40 "Dry Shielded Canister Transfer from Transfer Cask to HSM," Revision 4, (c) Radiological Protection Job Plan for ISFSI, Revision 0

Category: Radiation Protection **Topic:** Dose Rates During First Cask Loading

Reference: N/A

Requirement: Document dose rates during various work activities for the loading of the first canister and assess radiological controls used to keep doses low.

Observation: Good radiological controls were used throughout the loading of the first canister, as observed by the NRC inspectors. Certain areas on the refueling floor were designated as low dose areas. Health physics personnel were vigilant in keeping workers in those areas unless they were performing work. Health physics personnel were constantly cleaning the floors with cloths and checking for contamination. Entry into the work area around the cask was roped off and appropriate protective clothing requirements were enforced throughout the loading campaign. Health physics personnel were knowledgeable in practices to control dose and contamination to personnel and were actively involved with the work to ensure everyone was being protected. All health physics personnel interviewed during the work activities on the refueling floor by the NRC inspector had previous experience with other ISFSI projects. RWP/SWP 2010-037 and 2010-0114 were used to provide radiological controls during the work activities. Dosimetry included TLDs and alarming dosimeters, including alarming dosimeters capable of measuring neutrons. Health physics personnel constantly monitored dose rates during work activities. Some of the measured dose rates were: (a) 1-5 mrem/hr on the refueling floor in the ISFSI assigned work areas with most levels around 1 mrem/hr while the canister was in the spent fuel pool, (b) 70 mrem/hr beta/gamma and zero mrem/hr neutron on the top of the canister shield plug as the canister was coming out of the spent fuel pool, (c) 200 mrem/hr beta/gamma and 20 mrem/hr neutron on top of the canister shield plug after the canister was set in the cask washdown area and 1100 gallons (2/3) of water were removed from inside the canister prior to welding, leaving approximately 500 gallons in the canister, (d) 120 mrem/hr beta/gamma and 20 mrem/hr neutron after the inner lid was placed on the canister prior to welding, (e) 2-6 mrem/hr beta/gamma on the work platform during welding, (f) 8 mrem/hr beta/gamma and 4 mrem/hr neutron three feet from the side of the loaded transfer cask with all water removed from the canister and the transfer cask neutron shield full of water.

Airborne levels prior to the canister being removed from the spent fuel pool were from 2.65 x 10(-11) microcuries/cc to 2.28 x 10(-10) microcuries/cc. An Eberline AMS4 air monitor was being used with a calibration date of July 27, 2010 and a calibration due date of January 2011. Variances in radioactive air concentrations did not change

significantly throughout the loading campaign, with most increases attributable to increased radon levels in the building similar to what was being experienced throughout the plant from day-to-day.

Documents Reviewed: (a) Radiation Work Permit (RWP)/Special Work Permit (SWP) Authorization 2010-037 "ISFSI Project," dated June 10, 2010, (b) Radiation Work Permit (RWP)/Special Work Permit (SWP) Authorization 2010-114 "ISFSI Project SWP Areas," dated August 18, 2010

Category: Radiation Protection **Topic:** Evaluation of Effluent/Direct Radiation

Reference: 10 CFR 72.212(b)(2)(i)(C) & 10 CFR 72.104(a) Published 2010

Requirement: The general licensee shall perform a written evaluation prior to use that establishes that the requirements of 10 CFR 72.104 "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI" have been met. 10 CFR 72.104 requires that the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ during normal operations and anticipated occurrences.

Observation: The dose at the owner controlled area boundary from the Cooper ISFSI, based on 52 horizontal storage modules in a 2 x 26 back-to-back array loaded with 61BT canisters, was calculated to be below the 25 mrem/yr limit in 10 CFR 72.104. The thyroid dose and the critical organ dose were not applicable, since the NUHOMS cask system was a welded, leak tight system. The controlled area boundary's closest point was 800 meters to the north of the ISFSI. The NUHOMS Updated Final Safety Analysis Report (UFSAR), Section K.10.2 "Offsite Dose Calculations" calculated doses for normal operations for a 2 x 10 back-to-back array of twenty horizontal storage modules. This back-to-back array configuration would be similar to that used at Cooper. The computer code MCNP4 "Monte Carlo Neutron and Photon Transport Code System" (Oct. 1991) was used to calculate the doses. The source term was discussed in the UFSAR, Section K.5 "Shielding Evaluation." The General Electric (GE) 7 x 7 GE2/3 assembly design was used as the bounding source term because it had the highest initial heavy metal loading as compared to the other fuel assemblies allowed for storage in the 61BT canister. Four combinations of burnup, enrichment, and cooling time were considered in the calculations. These were: (a) 27 Gigawatt Days/Metric Ton Uranium (GWd/MTU), 2.00 wt % U-235, 5 year cooled, (b) 35 GWd/MTU, 2.65 wt % U-235, 8 year cooled, (c) 37.2 GWd/MTU, 3.38 wt % U-235, 6.5 years cooled, and (d) 40 GWd/MTU, 3.4 wt % U-235, 10 year cooled. These fuel specifications bounded the source terms for the spent fuel allowed for storage in the 61BT canister including the GE 8 x 8 fuel used at Cooper. Using the source term in Section K.5, calculations were performed for varying distances out to 600 meters in Section K.10.2 for the 2 x 10 array of twenty horizontal storage modules placed back-to-back. The calculations assumed 100% occupancy for 365 days. Table K.10-2 "Total Annual Exposure" provided the results of the calculations. Calculations were provided for both the front and the side of the array. The front of the array had the highest exposure levels. At 600 meters, the annual dose was calculated to be 10 mrem/yr. The dose dropped off significantly with distance from the ISFSI. For example, at 100 meters, the dose was 5,017 mrem/yr and at 300 meters the dose was 213 mrem/yr. For anticipated occurrences, UFSAR Section K.11.1 "Off-Normal Operations" reviewed the events that were not likely to occur on a regular basis, but could be

expected to occur with moderate frequency or on an order of once during a calendar year. None of the off-normal events resulted in additional exposure at the owner controlled area.

Cooper provided site specific dose calculations for their ISFSI consistent with the modeling in Section K of the UFSAR. Report NAI-1313-002 and Calculation 11301.0503 provided dose calculations for a 2 x 26 array of fifty-two horizontal storage modules (HSM-H) containing 61BT canisters and a discussion of the dose at the owner controlled area boundary. The horizontal storage module HSM-H is identical to the HSM-202 used at Cooper. The calculations used the MCNP/MCNPX Monte Carlo N-Particle Transport Code with MCNP51.40 and MCNPX 2.5.0. Skyshine was included in the calculations. Flux-to-dose conversion factors were used from the American National Standards Institute document ANSI/ANS 6.6.1 "American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from Light Water Reactor Nuclear Power Plants (1977)." Calculation 11301.0503, Table 2 "HSM Surface Average Dose Rate" provided the calculated dose from a single horizontal storage module. The surface dose rate on the front of the horizontal storage module was approximately 10 mrem/hr. The dose rate on the sides was approximately 0.2 mrem/hr. In comparison, the first canister loaded into HSM-1A (a corner location) at Cooper had a side dose rate of 0.2 mrem/hr and a front dose rate less than 1 mrem/hr. The lower front dose was reflective of the lower heat load of the first Cooper canister (11.3256 kW) compared to the maximum allowed for the HSM-H design of 40.8 kW listed in the Certificate of Compliance, Section 3.b "Cask Description." [Note that the 61BT canister is limited by Certificate of Compliance Table 1-1c "BWR Fuel Specifications for Fuel to be Stored in the Standardized NUHOMS 61BT DSC" to 300 watts per assembly. For 61 assemblies this totals 61 x 300 = 18.3 kW]. Future loadings will add to the front dose rate, but the side dose rate will remain approximately the same due to the end canister acting as a shield to the other canisters placed beside it. The dose rates from the horizontal storage modules were found in Calculation11301.0503 to be dominated beyond 400 meters by skyshine. Section 7 "Results" stated that the differences between the front, corner, and side dose values reduced with increasing distance resulting in similar doses at distances of 400 meters and beyond due primarily from skyshine. Therefore, doses would be expected to be symmetrical around the ISFSI at distances beyond 400 meters, or at least in the same order of magnitude. Section 7.2 "Total ISFSI Annual Exposures Due to 2 x 26 Array of HSM-H" provided several tables of results. Table 7 "Total ISFSI Annual Exposures at Different Distances from Long Side at the 2 x 4 Array Mid Point (2 x 26 Array Configuration)" provided the highest annual dose in comparison to the other tables for the 800 meter distance of 0.18 mrem/yr. However, this direction related to the long side of the array, which for Cooper was facing the owner controlled area such that the nearest boundary was 1,200 meters away. Using this longer distance dropped the dose to less than 0.4 mrem/yr. The table that reflected the shortest distance to the owner controlled area of 800 meters was Table 9 "Total ISFSI Annual Exposure at Different Distances from Short Side of the ISFSI" which listed a dose of 0.07 mrem/yr at 800 meters from a fully loaded ISFSI with a 2 x 26 array.

To meet the 10 CFR 72.104(a) dose limits, 10 CFR 72.104(a)(3) required any other radiation from uranium fuel cycle operations within the region to be added to the ISFSI dose. This would include the Cooper Nuclear Station. Radiological data from the 2006

through 2009 annual radiological environmental reports were used to develop an average offsite dose rate value. The average of these four years included a direct dose rate component of 0.7 mrem/yr, an annual dose rate from liquid effluents of 0.32 mrem/yr, and a dose rate from airborne releases of 0.04 mrem/yr. Adding the contributions from the nuclear plant to the ISFSI dose of 0.07 mrem/yr resulted in a total dose of 1.13 mrem/yr at the 800 meter owner controlled area. The dose for thyroid from the nuclear plant operations was 0.14 mrem/yr, well below the 75 mrem/yr limit. The dose to other critical organs was 1.47 mrem/yr, well below the 25 mrem/yr limit. These doses were discussed in the 72.212 Evaluation Report, Section 8.0 "10 CFR 72.212(b)(2)(i)(C) Radioactive Materials in Effluents and Direct Radiation."

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10, (d) Numerical Applications, Inc. Calculation NAI-1313-002 "Cooper Station ISFSI Offsite Accident Dose and Onsite Occupational Dose for 2 x 26 NUHOMS Facility," Revision 0, (e) Engineering Design Calculation (NEDC) 09-055 "CNS Review of Transnuclear Report NAI-1313-002," Revision 0, (f) AREVA/Transnuclear Calculation 11301.0503 "Far Field Dose Rates for Cooper Station ISFSI Comprised with HSM-H Loaded with NUHOMS -61BT Canister," Revision 0, (g) Engineering Design Calculation (NEDC) 09-057 "CNS Review of Transnuclear Calculation 11301.0503, Revision 0

Category: Radiation Protection **Topic:** HSM Dose Rates

Reference: CoC 1004, Tech Spec 1.2.7 Amendment 9

Requirement: When loaded with a 61BT canister, HSM dose rates are limited to: (a) 400 mrem/hour 3 feet from the horizontal storage module surface; (b) 100 mrem/hour on the door centerline; and (c) 20 mrem/hour on the end shield wall exterior.

Observation: Horizontal storage module dose rates after loading the first cask complied with Technical Specification 1.2.7 limits. Actual values after loading of the first canister into HSM-1A were less than 1 mrem/hr at three feet from the surface, 0.1 mrem/hr at the outside door centerline, and 0.2 mrem/hr on the end shield wall exterior. All dose rates were from gamma radiation. A neutron dose of 0.2 mrem/hr was detected on the shield door. Dose rates as high as 2,000 mrem/hr contact and 600 mrem/hr at 30 centimeters (cm) were measured on the canister at the opening to the horizontal storage module during insertion. Once inserted, the reading at the plane of the opening to the horizontal storage module was 600 mrem/hr and 100 mrem/hr at 30 cm. The vent screens on the bottom of the horizontal storage module read 25 mrem/hr on contact and 10 mrem/hr at 30 cm. Verification of compliance with the technical specification limits was incorporated into Step 14.18 of Procedure 10.40 and required sign-off by radiation protection and the cask loading supervisor.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.40 "Dry Shielded Canister Transfer from Transfer Cask to HSM," Revision 4

Category: Radiation Protection **Topic:** Neutron Dosimetry

Reference: UFSAR 1004, Sections 7.2.1 and 7.2.3 Revision 10

Requirement: Neutron sources are based on spontaneous fission contributions from six nuclides (predominantly Cm-242, Cm-244, and Cm-246) and (alpha, neutron) reactions due to eight alpha emitters (predominately Pu-238, Cm-242 and Cm-244). The total neutron source strength for BWR fuel is 1.01 x 10(8) neutrons per second per fuel assembly. The primary neutron source is the spontaneous fission of Cm-244 which represents 85% of the total neutron sources. Table 7.2-2 provides the neutron energy spectrum for BWR fuel.

Observation: Neutron survey instruments used for the cask loading activities adequately monitored the neutron dose to workers and considered the neutron spectrums that would be present during the various work activities. When water was in the canister or the canister was in the horizontal storage module, the neutron spectrum was moderated and the normal dosimetry utilized at the plant adequately monitored the neutron dose. However, when water was drained from the canister, the neutron spectrum resembled that of an unmoderated neutron source. For neutron dose rate measurements, the licensee utilized the Far West Technology REM-500, a tissue equivalent proportional chamber. The instruments used were calibrated by Far West Technology to Bare Cf-252, using a National Institute of Standards and Technology (NIST) traceable source. In addition, the licensee obtained and utilized a NIST study of exposure of the REM-500 to moderated and bare neutron sources. For the NIST study, the Far West Technology REM-500 instruments were mounted on a stand and measurements made at various distances from the sources in order to obtain different dose equivalent rates. For the REM-500, the NIST study determined a calibration factor which was the factor by which the rem-meter reading should be multiplied to get the true dose equivalent rate in mrem/hr. The NIST study determined that for both bare and moderated californium sources, the instrument can be considered to have the same calibration factor. Thus, the NIST study concluded that the REM-500 responded similarly to both moderated and unmoderated neutron spectra and therefore would provide for adequate assessment of neutron exposures for all situations that would occur during the loading campaign. The table provided in the NIST study gave a mean calibration factor value of 1.24 +/- 8% for the moderated californium source and 1.20 +/- 8% for the bare californium source. During the first canister loading, two Rem-500 neutron survey meters, Serial # 396 and # 417 were available at the cask work area. Both had been calibrated May 17, 2010 with calibration due dates of May 17, 2011.

For personnel monitoring, a CR-39 chip had been included in the personnel dosimeter of legal record to be worn when neutron fields may be present. The CR-39 chip has a relatively flat response to the energy spectrum over the energy range expected around the casks. Neutron sensitive alarming electronic dosimeters (Mirion DMC 2000GN) were also being used. During the loading of the first cask, only two individuals showed any neutron dose. One individual received 20 mrem and one individual showed 30 mrem on their dosimeter of legal record. Numerous other individuals had shown neutron doses on their electronic dosimeters, with the highest values at 50 mrem, but no dose was collected on their dosimeter of legal record. The individual that received 30 mrem on his dosimeter of legal record had a 34 mrem estimate from the electronic dosimeter. The individual that had 20 mrem on his dosimeter of legal record showed 2 mrem on his

electronic dosimeter.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Report of Test, Far West Technology REM-500, NIST Test No. 2832, US Department of Commerce, National Institute of Standards and Technology (NIST), Gaithersburg, MD., dated March 8, 1992, (c) Health Physics Instruments Rem 500 Calibration Sheet for Serial No. 396, calibrated May 17, 2010

Category: Radiation Protection **Topic:** Radioactive Gas Sample Prior to Unloading

Reference: UFSAR 1004, Section K.8.2.2.19. Revision 10

Requirement: If fuel needs to be removed from the canister, precautions must be taken against the potential for the presence of damaged or oxidized fuel and to prevent radiological exposure to personnel during this operation. A sampling of the atmosphere within the canister will be taken prior to inspection or removal of the fuel.

Observation: Procedures for cutting open a canister and removing the fuel included a step to sample the atmosphere prior to the lid removal to prevent unexpected radiological exposures to workers during the lid removal process. Procedure 10.38.1 incorporated the requirement to sample the atmosphere of the canister prior to unsealing. Section 2 "Precautions and Limitations" of the procedure provided precautions applicable to this effort and noted that fission product gases, in particular Kr-85, may be present in the canister. Sampling was performed in Steps 5.12 through 5.23. Step 5.24 directed chemistry to analysis the gas sample and report back if fission gases were present and to verify helium was still present. If fission gases were present, a caution stated that appropriate filtering should be initiated to preclude uncontrolled radioactive particulate release from the canister purge valves. A caution noted that damaged fuel may release significant radioactive gases into the canister and that area radiation levels will thus increase during purging of the canister. Contact dose rates on the gas sample collected could reach 1.6 rem/hr. Section 7 "DSC Inner Top Cover Removal" outlined precautions to be taken, including stoppage of cutting operations, should hydrogen concentrations inside the canister exceed 2.4%. Attachment 3 "Gas Sample Bottle" presented details of the gas sample bottle that will be utilized. Chemistry procedures provided instructions for performing the gas analysis to quantify fission products that may be present in the canister.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Nuclear Performance Procedure 10.38.1 "Dry Shielded Canister Unsealing," Revision 1

Category: Radiation Protection **Topic:** Transfer Cask Dose Rates

Reference: CoC 1004, Tech Spec 1.2.11 Amendment 9

Requirement: When containing a loaded 61BT canister, the transfer cask dose rates shall be limited to 200 mrem/hr at 3 feet from the cask surface with water in the canister cavity and 500 mrem/hr at 3 feet without water in the canister cavity. The dose rates should be determined as soon as possible after the transfer cask is removed from the spent fuel pool.

Observation: The dose rates on the transfer cask were within the Technical Specification 1.2.11 limits

for the first cask loaded. As the transfer cask was being removed from the spent fuel pool, a radiological survey was performed to measure the dose rate at the mid-plane of the cask using Procedure 10.38, Steps 4.44 thru 4.50. For this measurement, water was in the canister. The highest dose rate at three feet was 5 mrem/hr gamma and zero mrem/hr neutron. The dose rate (gamma plus neutron) met the 200 mrem/hr limit at three feet. After the canister had undergone final pump-down to remove all water prior to initial vacuum drying, a dose rate measurement was required by Step 7.22 to confirm compliance with the 500 mrem/hr limit of Technical Specification 1.2.11 with no water in the canister. The highest dose rate at three feet was 8 mrem/hr gamma and 4 mrem/hr neutron at the mid-plane of the cask. The heat load for the first canister was 11.3256 kW.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Nuclear Performance Procedure 10.38 "Dry Shielded Canister Sealing," Revision 4

Category: <u>Records</u> **Topic:** <u>Cask Records</u>

Reference: 10 CFR 72.212(b)(8) Published 2010

Requirement: The licensee shall accurately maintain the records provided by the cask supplier for each cask that shows, in addition to the information provided by the cask vendor, the following: (a) the name and address of the cask vendor, (b) the listing of the spent fuel stored in the cask and (c) any maintenance performed on the cask. This record must include sufficient information to furnish documentary evidence that any testing and maintenance of the cask has been conducted under an NRC approved QA plan.

Observation: The licensee had an established filing system for maintaining the records provided by the cask supplier for each cask. Initially, these records were kept in hard copy in the ISFSI trailer with an electronic version maintained on a server in a remote location. Once the first loading campaign was completed, the records were added to the licensee's permanent records system. Procedure 1.9 identified records pertaining to dry fuel storage as lifetime quality assurance (QA) records and specifically stated that the retention period shall be for as long as spent fuel was stored at the ISFSI and for a period of 5 years after the material was disposed of or transferred out of the ISFSI. The procedure also contained instructions to identify quality records generated in support of 10 CFR Part 72 licensing requirements as "ISFSI Records" by use of a stamp on the document and by identifying the record as an "ISFSI" type record on the record transmittal form. After the canisters were loaded, all procedures completed during loading of the canister, including the records showing which fuel assemblies had been placed in the canister and their location, were required to be added to the permanent records.

The Certificate of Conformance for canister Serial Number CNS61B-008-A, provided by Transnuclear, Inc., was reviewed. It included the name and address of the cask vendor, a list of any nonconformances from requirements of Transnuclear drawings or specifications, and a statement that the canister was designed, fabricated, tested, and repaired in accordance with the Transnuclear QA program for activities conducted under 10 CFR Parts 71 and 72.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Site Service Procedure 1.9 "Control and Retention of Records," Revision 50, (c) Certificate of Conformance, DSC Serial Number CNS61B-008-A

Category: Records **Topic:** Maintaining a Copy of the CoC and Documents

Reference: 10 CFR 72.212(b)(7) Published 2010

Requirement: The general licensee shall maintain a copy of the Certificate of Compliance (CoC) and documents referenced in the certificate.

Observation: The licensee hwas maintaining copies of the Certificate of Compliance, Technical Specifications, Final Safety Analysis Report, and NRC Safety Evaluation Report. 10 CFR 72.212(b)(7) also required that the licensee maintain a copy of all documents referenced in the Certificate of Compliance, for each cask model used for storage of spent fuel. The Technical Specifications in the Certificate of Compliance were reviewed and several documents referenced in it were identified. These included NRC guidance documents (NUREGs and Interim Staff Guidance), American National Standards Institute (ANSI) documents, and technical reports from a national laboratory. Attempts made to access these documents at the licensee's technical library or through their online access were unsuccessful. This was brought to the attention of the licensee during the inspection. The licensee generated Open Item No. 109 to capture this issue and successfully obtained a copy of the missing documents and placed them in the licensee's technical library database.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 and Attachment A "Technical Specifications," Amendment No. 8, (c) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUH-003), Revision 10, (d) Safety Evaluation Report (SER), Docket No. 72-1004 "Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," Amendment No. 9

Category: Records **Topic:** Notice of Initial Loading

Reference: 10 CFR 72.212(b)(1)(i) Published 2010

Requirement: The general licensee shall notify the NRC at least 90 days prior to first storage of spent fuel. The notice may be in the form of a letter, but must contain the licensee's name, address, reactor license and docket number, and the name and means of contacting a person responsible for providing additional information concerning spent fuel under this general license. A copy of the submittal must be sent to the administrator of the appropriate NRC regional office.

Observation: Nebraska Public Power District complied with the 90-day notification requirement on January 8, 2009. A letter was sent to the NRC informing the agency of the plans to load spent fuel under a general license at the Cooper Nuclear Station. The letter included the required information specified in 10 CFR 72.212(b)(1)(i). NRC Region IV was copied on the letter.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Letter (NLS2008107) from David W. Van Der Kamp, Nebraska Public Power District to NRC Document Control Desk entitled "90-Day Notification Pursuant to 10 CFR 72.212(b)(1)(i) of Intent to Load Spent Fuel Under a General License - Cooper Nuclear Station Docket No. 50-298, DPR-46," dated January 8, 2009 [NRC ADAMS Accession No. ML090230634]

Category: Records **Topic:** Record Retention for 72.212 Analysis

Reference: 10 CFR 72.212(b)(2)(i)(C) Published 2010

Requirement: A copy of the 10 CFR 72.212 analysis shall be retained until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

Observation: The requirement to retain the Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," in compliance with 10 CFR 72.212(b)(2)(i)(C) had been incorporated into Site Services Procedure 1.9 and the Records Retention Schedule. Procedure 1.9, Step 2.6.4 required Cooper Nuclear Station's dry cask storage records to be maintained in accordance with 10 CFR Part 72 with a retention period of five years after the material was disposed of or transferred from the ISFSI.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station (CNS) Records Retention Schedule, Revision 33, (c) Site Services Procedure 1.9, "Control and Retention of Records" Revision 50

Category: Records **Topic:** Registration of Casks with NRC

Reference: 10 CFR 72.212(b)(1)(ii) Published 2010

Requirement: The general licensee shall register the use of each cask with the NRC no later than 30 days after using the cask to store spent fuel.

Observation: Notification to the NRC within the 30 day requirement was completed for the first cask loaded. The cask was loaded into the horizontal storage module on October 21, 2010. The 30-day letter was issued to the NRC on November 15, 2010. The requirement to notify the NRC when a canister was placed into the horizontal storage module was included in Procedure 10.40, Step 14.13. This step required completion of Attachment 9 "Licensing Department Notification of DSC Use" and notification of the licensing manager that the cask loading was complete. Information required in the attachment included both the canister and horizontal storage module serial numbers thus meeting the NRC reporting requirement as specified by 10 CFR 72.212 (b)(1)(ii). Attachment 9 directed that a licensing order be issued to track the required 30 day NRC notification per 10CFR72.212(b)(1)(ii).

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) Nuclear Performance Procedure 10.40 "Dry Shielded Canister Transfer from Transfer Cask to HSM," Revision 4, (d) Letter (NLS2010099) from David W. Van Der Kamp, Nebraska Public Power District to NRC Document Control Desk entitled "Thirty-Day Notification Pursuant to 10 CFR 72.212, Condition of General License Issued Under 72.210 for Storage of Spent Fuel at Cooper Nuclear Station, Docket 50-298, DPR-46, Cooper Nuclear Station ISFSI, Docket No. 72-66" dated November 15, 2010 [NRC ADAMS

Accession No. ML103270518]

Category: Slings **Topic:** Sling Heavy Load Requirements

Reference: NUREG 0612, Section 5.1.6 (1) (b) Issued July 1980

Requirement: One of the following should be satisfied unless the effects of a drop of the particular load have been analyzed and shown to satisfy the criteria of Section 5.1 of NUREG 0612: (i) dual or redundant slings or lifting devices should be used such that a single component failure or malfunction in the sling will not result in an uncontrolled lowering of the load, or (ii) the load rating of the sling should be twice the sum of the static and dynamic loads.

Observation: The licensee complied with the NUREG requirement. During the load handling operations for a canister using the NUHOMS system, no lifts of the loaded canister involved the use of slings. Only one lift used slings that held a load above the fuel assemblies. This lift was the lifting of the shield plug, which was installed on top of the canister inside the spent fuel pool after the spent fuel had been loaded into the canister. The shield plug rigging arrangement per Procedure 10.37 Attachment 3 "Shield Plug Lift Assembly" utilized four wire rope slings. Step 3.1.19.1 of Procedure 10.37 defined the rating for the slings and the attachments. Each of the four wire rope slings were designated to be rated at a minimum of 22,800 lbs. Eight steel shackles were used (two per sling), each rated at 8,000 lbs minimum. Eight hoist rings were used (two per sling), each rated at 5,000 lbs minimum. Four stainless steel jaw to jaw turnbuckles were used (one per sling), each rated at 5,200 lbs. The shield plug weighed approximately 7,016 lbs per licensee Calculation 08-042. Failure of one out of the four slings would not cause an uncontrolled lowering of the shield plug. With four slings, each carried 7,016 lbs/4 = 1,754 lbs. The hoist rings, as the weakest link, were rated at 5,000 lbs. This would be well above the required factor of two required by the NUREG.

Documents Reviewed: (a) NUREG 0612 “Control of Heavy Loads at Nuclear Power Plants,” issued July 1980, (b) Maintenance Procedure 7.1.8 "Rigging and Lifting at CNS," Revision 26, (c) Calculation 08-042 "Transnuclear Transfer Cask (TC) and Dry Shielded Canister (DSC) Weights for Various Spent Fuel Loading Configurations," Revision 0, (d) Nuclear Performance Procedure 10.37 "Dry Shielded Canister Loading," Revision 5

Category: Slings **Topic:** Synthetic Sling Removal From Service

Reference: ASME B30.9 (1971) Section 9-5.6.2 Revision 1971

Requirement: A synthetic webbing sling shall be removed from service if any of the following conditions are present: (a) acid or caustic burns, (b) melting or charring of any part of the surface, (c) snags, punctures, tears, cuts, (d) broken or worn stitching, (e) wear or elongation exceeding the amount recommended by the manufacturer, (f) distortion of fittings, and (g) other apparent defects which cause doubt as to the strength of the sling should be referred to the manufacturer for determination.

Observation: Procedure 7.2.76, Section 11 "Synthetic Web Slings, Examination and Testing" and Section 12 "Periodic (Annual) Examination of Synthetic Web Slings" provided controls to ensure that damaged slings were removed from service. Step 11.1 required frequent examinations of all slings by the rigger, the person handling the sling, or the tool crib person prior-to-use each day the sling is used. The inspection consisted of a visual

observation for gross damage. The sling was required to be removed from service if any of the following were visible: (a) acid or caustic burns, (b) melting or charring of any part of the sling, (c) holes, tears, cuts, or snags, (d) broken or worn stitching in load-bearing splices, (e) excessive abrasive wear, (f) knots in any part of the sling, (g) excessive pitting or corrosion, (h) cracked, distorted, or broken fittings, (i) discoloration of sling material, and (j) missing ID tag which was required to show: CNS number, rated load capacity, and date of the last periodic examination which should have occurred within the previous year. If no tag was found, the sling was to be returned to the tool crib and not used.

Section 12 provided requirements for the annual inspection. Step 12.1 required an end-to-end examination of synthetic web slings at least annually and required a record of the examination to provide the basis for a continuing evaluation of the equipment. The examination was performed per Attachment 9 "Synthetic Web Sling Annual Examination Report" with particular attention to the following: (a) acid or caustic burns to the sling/web material, (b) broken or torn stitching, (c) excessive wear due to abrasion, (d) knots in sling/web material, (e) cracked or damaged fittings, (f) any other visible damage, and (g) discoloration of sling material. A durable tag with the date of the annual examination attached to the sling in such a manner that it would not interfere with the operation of the sling was also required.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) B30.9 “Slings,” Revision 1971, (b) Maintenance Procedure 7.2.76 "Sling, Fall Protection Harness/Lanyard Examination, Maintenance, and Testing," Revision 8

Category: Special Lifting Devices **Topic:** Lift Yoke Load Test

Reference: UFSAR 1004, Section 3.4.4.1 and 4.2.3.3 Revision 10

Requirement: The yoke is designed and fabricated to meet the requirements of ANSI N14.6 (1986). The test load for the yoke is 300% of the design load, with annual dimensional and liquid penetrant or magnetic particle inspections to meet the ANSI N14.6 requirements.

Observation: The licensee's OS197H transfer cask lift yoke had a capacity of 110 tons and was designed to meet the requirements of the American National Standards Institute (ANSI) N14.6 guidance. The lift yoke was tested to 300% of the rated load per Transnuclear Document Number 94010-T-003 on August 9, 2010. The results of the load test were recorded on Load Test Procedure Data Sheet No. 1008091500. The lift yoke was tested in an apparatus that used pressurized pistons to apply the load to the lift yoke. The test rig was pressurized to 6,500 psi which translated to a load of approximately 680,000 lbs of force (340 tons). This test load was greater than the required 300% load test of 110 tons times three equals 330 tons. The load was held for ten minutes without a drop in pressure. A visual inspection was conducted after the test which found no issues.

The NRC inspectors identified that the annual requirement specified in the NUHOMS Updated Final Safety Analysis Report (UFSAR), Section 4.2.3.3 required a 300% load test and visual examination plus a liquid penetrant or magnetic particle inspection to meet the ANSI N14.6 requirement. However, the wording in Section 6.3.1 of ANSI N14.6 allowed for either a 300% load test and visual examination or a liquid penetrant or magnetic particle inspection. As such, Cooper met the requirement of the ANSI N14.6

requirement but did not meet the UFSAR requirement. Contact with Transnuclear, Inc. determined that the UFSAR statement was in error. Cooper issued Condition Report CR-CNS-2010-06815 to document this issue. Transnuclear Inc. initiated FSAR Change Notice FCN-721004-868 to document the clarification for a future revision to the UFSAR.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Transnuclear Document Number 94010-T-003 "Load Test Procedure for OS197H Transfer Cask Lifting Yoke 110 Ton Capacity", Revision 0, Attachment A, Data Sheet No. 1008091500, dated August 12, 2010, (c) American National Standards Institute N14.6 "Special Lifting Devices for Shipping Containers Weighing 10,000 lbs or More," Revision 1986, (d) Cooper Condition Report CR-CNS-2010-06815 "TN UFSAR Inconsistency with ANSI N14.6," dated September 17, 2010, (e) TN FSAR Change Notice FCN-721004-868, dated September 22, 2010

Category: Training **Topic:** Certification of Personnel

Reference: 10 CFR 72.190 Published 2010

Requirement: Operations of equipment and controls that have been identified as important-to-safety in the SAR and in the license must be limited to trained and certified personnel or be under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of equipment and controls that are important-to-safety must also be certified in such operations.

Observation: The licensee's certification program included requirements for the selection, training, and qualification of station personnel. The licensee's program for selection and training of station personnel was delineated in Administrative Procedure 0.17. The procedure also explained the process for removal of an individual's qualification status. Qualification cards were used by the licensee to document successful completion of worker training and qualifications. Several qualification cards were reviewed to determine if they adequately covered the required work activities such that an individual who had completed the qualification card would have an adequate understanding of the requirements and be knowledgeable to perform the assigned work activity. Qualification Card SKL8280010/40702 "MEC Perform DSC-TC Preparation for Fuel Loading" documented completion of the training related to the storage and handling of the transfer cask and canister including the requirements to prepare the canister and transfer cask for fuel loading. Qualification Card SKL8280020/40703 "MEC Perform DSC Sealing Operations" documented completion of activities associated with canister sealing operations. Qualification Card SKL8280030/40704 "MEC Perform DSC-TC Transfer To or From the HSM" documented completion of activities associated with moving the transfer cask on the transport trailer to and from the ISFSI pad.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) Administrative Procedure 0.17 "Selection and Training of Station Personnel," Revision 54, (c) Qualification Card "MEC Perform DSC-TC Preparation for Fuel Loading (SKL8280010/40702)," Revision 00, (d) Qualification Card "MEC Perform DSC Sealing Operations (SKL8280020/40703), Revision 00, (e) Qualification Card "MEC Perform DSC-TC Transfer To or From the HSM (SKL8280030/40704)," Revision 00

Category: <u>Training</u> **Topic:** <u>Health Requirement for Certified Personnel</u>

Reference: 10 CFR 72.194 Published 2010

Requirement: The physical condition and the general health of personnel certified for the operation of equipment and controls that are important-to-safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important-to-safety. These conditions need not categorically disqualify a person if appropriate provisions are made to accommodate such defect.

Observation: The licensee's program included the requirement to evaluate the general health of individuals assigned to operate equipment important-to-safety. This evaluation was performed by a medical doctor and consisted of the physical exam used to qualify an individual to wear a respirator. Once the exam was completed, it was documented in the training computer system as GEN0020201. The training matrix, which listed persons qualified for the ISFSI project, included the record that the individual had successfully completed the medical exam. The mechanical maintenance staff was the primary organization assigned to operate the important-to-safety equipment. The qualification cards (SKL8280010, SKL8280020, and SKL8280030) all included the prerequisite of a physical exam. The training matrix was the tool used to verify all personnel assigned to the first cask loading had completed the physical requirements. To be identified in the training matrix as certified to perform the specific task, the individual must have satisfied the health requirement. The licensee had not identified anyone in the certification program that required special provisions to meet the certification requirements.

Documents Reviewed: (a) Code of Federal Regulations (CFR), Title 10 "Energy," published 2010, (b) GEN00220201 "Physical Qualifications," Revision 02.02, (c) Qualification Card "MEC Perform DSC-TC Preparation for Fuel Loading (SKL8280010/40702)," Revision 00, (d) Qualification Card "MEC Perform DSC Sealing Operations (SKL8280020/40703)," Revision 00, (e) Qualification Card "MEC Perform DSC-TC Transfer To or From the HSM (SKL8280030/40704)," Revision 00

Category: <u>Training</u> **Topic:** <u>Required Training for ISFSI Staff</u>

Reference: CoC 1004 Tech Spec 1.1.5; UFSAR 1004, Sect. 9.3 Amendment 9/Rev. 10

Requirement: Generalized training should be provided to ISFSI personnel in the applicable regulations and standards and the engineering principles of passive cooling, radiological shielding, and structural characteristics of the ISFSI. Detail training shall be provided for canister preparation and handling, fuel loading, transfer cask preparation and handling, and transfer trailer loading.

Observation: General training was provided to ISFSI personnel on the applicable regulations and standards and the engineering principles of passive cooling, radiological shielding, structural characteristics of the ISFSI, and the process of performing a cask loading campaign. This was presented in the first training module entitled "ISFSI System Overview." Additional specialized training was then provided for specific groups such as engineering personnel, radiation protection personnel, quality control personnel, and mechanical engineering personnel assigned to cask loading operations. The training

provided to mechanical maintenance personnel via the MEC8280000 series (i.e., MEC3001009, MEC3001010, MEC3001011, and MEC3001012) instructed the trainee on the standardized NUHOMS design; ISFSI facility design; the certificate of compliance (CoC); and fuel loading, transfer cask handling, and canister transfer. Training Module MEC3001009 covered: NUHOMS® system operations, ISFSI facility design, and certification of compliance (CoC) overview. Training Module MEC3001010 covered canister and transfer cask offloading, handling, and storage. Training Module MEC3001011 covered canister loading preparation and sealing operations [e.g., loading, draining and backfilling the canister, operation of the vacuum drying skid (VDS) and installation of the automated welding system (AWS)]. Training Module MEC3001012 covered fuel loading, transfer cask handling, and canister (DSC) transfer procedures. Also presented in Training Module MEC3001012 were the procedures used to transport the canister to the horizontal storage module; precautions and limitations for material movement; rigging hardware requirements; and an outline for the steps to transfer the dry shielded canister/transfer cask to the transport trailer.

Documents Reviewed: (a) Certificate of Compliance No. 1004 for the Transnuclear, Inc. Standardized NUHOMS® Horizontal Modular Storage System," Amendment No. 9 [NRC ADAMS Accession No. ML071070570], (b) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (c) Training Module MEC3001009 "Independent Spent Fuel Storage Installation (ISFSI) Overview," Revision 0, (d) Training Module MEC3001010 "Dry Shielded Canister (DSC) & Transfer Cask Offloading, Handling & Storage," Revision 0, (e) Training Module MEC3001011 "DSC Loading Preparation & Sealing Operations," Revision 0, (f) Training Module MEC3001012 "DSC Transfer Operations & Transfer Cask Transit to HSM," Revision 0, (g) Training Module CNSESP00911 "CNS ISFSI Overview," Revision 0, (h) Training Module OTH0150906 "ISFSI System Overview," Revision 0, (i) Training Module OTH0150911 "ISFSI System Overview," Revision 1, (j) AREVA RAD9080401R01-PP "Dry Cask Storage - NUHOMS General Systems Overview," Revision 1

Category: Training **Topic:** Training for Health Physics Staff

Reference: UFSAR 1004, Sections 9.3.1.3 Revision 10

Requirement: Training should be provided to plant health physics personnel on applicable regulations and standards and in the engineering principles of passive cooling, radiological shielding, and structural characteristics of the ISFSI. Specific training should be provided in radiological shielding design of the system, particularly the DSC top shield plug, the transfer cask and the HSM.

Observation: Training was provided to plant and contractor health physics personnel that participated in the first loading campaign that satisfied the requirements of Technical Specification 1.1.5 "Training Module" and the NUHOMS Updated Final Safety Analysis Report (UFSAR), Section 9.3 "Training Program." The training provided was described in the 10 CFR 72.212 Evaluation Report, Section 13.2.3 "Training Program." A training needs analysis worksheet was completed which determined that there were no new tasks for radiological control technicians (RCTs). However, additional training was necessary for some of the unique requirements of the cask loading process. This was provided in the training that included Lesson Plan AREVA RAD9080401 "Dry Cask Storage -

NUHOMS General Systems Overview." Contractors were given credit for previous classroom training but were required to complete site specific "on the job" training. Dry runs were utilized for qualification of staff and contractors. Principles of a Systematic Approach to Training and an industry accepted accreditation were employed to ensure proper analysis of requirements and effectiveness of the ISFSI specific training.

Documents Reviewed: (a) Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (NUH-003), Revision 10, (b) Cooper Nuclear Station "10 CFR 72.212 Evaluation Report," Revision 0, (c) CNS Radiological Protection Job Plan for ISFSI, (d) AREVA RAD9080401R01-PP "Dry Cask Storage - NUHOMS General Systems Overview," Revision 1

Category: Welding **Topic:** Tack Welds

Reference: ASME Section III, Article NB-4231.1 Code Year 2001

Requirement: Tack welds used to secure alignment shall either be removed completely when they have served their purpose, or their stopping and starting ends shall be properly prepared by grinding or other suitable means so that they may be satisfactorily incorporated into the final weld. When tack welds are to become part of the finished weld, they shall be visually examined and defective tack welds shall be removed.

Observation: Procedure 06260-CNS-OPS-01 incorporated the requirement to prepare the tack welds for consumption into the final weld and to perform a visual examination of the tack welds. Step 8.6.4.C stated "If tack welds are to be consumed within the finished weld, all stop, start, uneven surfaces or if excessive proportionate to the joint design shall be ground and feathered to ensure proper incorporation. They shall be visually examined and free of defects. Defective tack welds shall be completely removed." Sign-off of the visual inspection was provided in Attachment 9.3 "Weld Data Sheet," for the tack welds on the inner top cover to shell, outer top cover to shell, siphon port and vent port. Typically, eight tack welds approximately 2" long were used on the lids. These tack welds were then incorporated into the weld.

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III "Rules for Construction of Nuclear Facility Components," 2001 Edition, (b) TriVis Procedure 06260-CNS-OPS-01 "Spent Fuel Cask Welding 61BT NUHOMS Canister," Revision 3, (c) TriVis Welding Procedure Specification WPS 06260-CNS-SS-8-A-TN "Weld Procedure Specification," Revision 2 and Supplemental Information

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Category: Welding **Topic:** Weld Lengths

Reference: ASME Section III, Article NF-5360 Code Year 2001

Requirement: For welds 3 inches and longer, weld lengths shorter than specified by more than 1/4 inch (6 mm) are unacceptable. For welds less than 3 inches long, weld lengths shorter than specified by more than 1/8 inch (3.2 mm) are unacceptable. Intermittent welds not spaced within 1 inch (25 mm) of the specified location are unacceptable.

Observation: No unacceptable welds were identified on the first canister. The welding for the inner and outer top covers were performed by an automatic welding machine that welded the entire length of the weld in a single pass for each of the welds performed (i.e. root pass

weld, intermediate welds and final weld.)

Documents Reviewed: (a) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III "Rules for Construction of Nuclear Facility Components," 2001 Edition, (b) TriVis Quality Procedure 06260-CNS-QP-9.201 "Visual Weld Examination of Dry Cask Assembly," Revision 6