March 6, 2007

MEMORANDUM TO: Michael Karmis, Safety Engineer
Rules, Inspections and Operations Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards

FROM: Jack W. Foster, Branch Chief */RA/*
Operating Experience and Generic Issues Branch
Operating Experience and Risk Analysis
Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research

SUBJECT: REQUESTED GENERIC ISSUE (GI)-203, POTENTIAL SAFETY ISSUES WITH CRANES THAT LIFT SPENT FUEL CASKS

The requested generic issue (GI) that you submitted to the Generic Issues Program (GIP) in an e-mail on December 21, 2006, described your concerns about potential safety issues with cranes used to lift spent fuel casks at nuclear power plants. On January 11, 2007, the GIP assigned your requested GI number 203. On January 11 and 18, 2007, Harold Vandermolen, Timothy Mitts, and I met with you and Mr. Robert Lewis to discuss your concerns with crane safety at nuclear power plants. The purpose of these meetings was to obtain a clear understanding of the scope of the requested GI by identifying specific concerns to include, and any to exclude, from consideration under the GIP. In a memorandum (ADAMS ML070230133), dated January 29, 2007, I provided you minutes from these meetings that presented the consensus conclusions and identified follow-up actions.

These meetings resulted in several follow-up actions described in the meeting minutes. One of these actions was for the GIP to: determine the suitability of your concerns for further consideration under the GIP per Management Directive (MD) 6.4, provide the basis for this determination, and provide you a formal response for the agreed upon four areas of concern described in the requested GI-203.

The GIP review of your concerns, which included input from NRR and NMSS management, determined they represent licensee compliance issues and as such are not suitable for further consideration under the GIP per Management Directive (MD) 6.4. The areas of your concern (summarized below) involve questions about the technical adequacy or programmatic effectiveness of existing regulatory programs and activities that are implemented through the reactor oversight process (ROP) as described in the enclosure. Accordingly, your concern has been entered into the ROP Feedback Program as ROP Feedback Form Item Number 60854-1-1113.

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(1) Cranes that do not conform to original design specifications: load drop analysis is not part of design basis, inadequate design basis documentation / information from parts vendors, and NRC inspection responsibilities for review of load drop analysis are unclear.

(2) Cranes that licensees modified without performing safety evaluation reviews required by 10 CFR 50.59, which may indicate some licensee's assume that these reviews are not required.

(3) Cranes that have had inadequate maintenance: overlooks important operating experience, invalidates single-failure proof capability, and tolerated by deficient NRC requirements for maintenance.

(4) Cranes that are not single-failure proof or lack credible validation for single-failure proof status have been used without adequate load path protection, or other mitigative measures, and this condition is not adequately considered in NRC's probabilistic risk assessment of crane events.

The regulatory issue summary (RIS) 2005-25: "Clarification of NRC Guidelines for Control of Heavy Loads," describes the agency's regulatory position with respect to the areas of concern identified in your requested GI-203. See:
http://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2005/ri200525.pdf

This RIS identifies the design and inspection bases for cranes having the potential to impact important to safety SSCs. This RIS addresses single-failure-proof cranes and includes guidelines for upgrading cranes to single-failure-proof status as well as for crane inspection, testing, and maintenance. This RIS also addresses conditions requiring a load drop analyses and provides guidance on load drop analysis assumptions and methods. The RIS incorporates operating experience (NUREG 1774, A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002, published July 2002), including results from staff work associated with generic safety issue (GSI)-186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," which form the risk-informed basis for the NRC using the RIS to clarify NRC guidelines for licensee's programs for control of heavy loads.

One possible net impact of your four areas of concern is that licensees have maintained inadequate control of changes to cranes over long periods of time. Sources of crane changes to consider include: design modifications, parts and information from vendors, aging, maintenance (or lack of thereof), and incorporation of lessons learned from operating experience. Potential consequences from licensees inadequate control of these changes may result in conditions where cranes no longer conform to original design specifications, conditions that invalidate load drop analysis, or conditions that invalidate single-failure-proof status of cranes. Licensee's control of design basis of cranes that provide important to safety functions is clearly within the purview of the ROP inspection procedures as described above. In addition, guidance from RIS 2005-25 describes the NRC's regulatory position that licensee's can maintain adequate defense-in-depth through: use of single-failure-proof cranes, providing various appropriate forms of load path protection, or performing adequate load drop analysis that demonstrates acceptable consequences. Thus, licensee's may choose among these alternatives to maintain adequate defense in depth (i.e., they are not specifically obligated to

maintain original design basis, to perform load drop analysis, or to provide load path protection for cranes providing important to safety functions). Again, these considerations are part of the existing regulatory framework and therefore are not appropriate for further evaluation under the GIP.

In conclusion, the four agreed upon areas of concern identified in your requested GI are covered under existing inspection procedures of the ROP. The enclosure provides a sample of inspection reports for licensee activities involving ISFSI, DCSS, and other heavy load campaigns as well as their interface with (and potential for adverse impact on) plant SSCs, including cranes used to lift spent fuel casks and potential adverse impact on plant programs. The guidance provided to licensees in RIS 2005-25 clearly describes the agency regulatory position for licensee's control of heavy loads programs covering the areas of your concerns. Therefore, the areas of your concern represent licensee compliance issues and as such are not suitable for further assessment under the GIP. That is, criteria for screening candidate generic issues described in MD 6.4 excludes issues that would be obviated by licensee compliance with existing rules, guidance, or programs.

Your concern has been entered into the ROP Feedback Program in accordance with Inspection Manual Chapter 0801 as ROP Feedback Form Item Number 60854-1-1113. See link to Status Reports of existing ROP Feedback where entries are listed by Inspection Procedure.

Please contact Timothy Mitts if you would like to discuss these conclusions or their basis.

Enclosure:
Inspection Procedures and Reports Applicable to Licensee's Cranes
That Lift Spent Fuel Casks and Other Heavy Loads

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Enclosure

Inspection Procedures and Reports Applicable to Licensee's Cranes That Lift Spent Fuel Casks and Other Heavy Loads

The NRC's reactor oversight process includes several inspection program manual chapters and inspection procedures that describe inspection requirements and inspection guidance for inspections of activities involving cranes. The inspection procedures cover cranes used to lift spent fuel casks and cranes used to lift other heavy loads that have a potential to adversely impact plant structures, systems and components (SSCs) that are important to safety. These inspection documents are publically available and may be used by licensees and NRC inspectors to prepare for and to perform these inspections. A sample of NRC inspection reports from 2001 through 2006, that document NRC's inspection of licensee's independent spent fuel storage installation (ISFSI), dry cask storage system (DCSS), and other heavy load campaigns are summarized below. These inspections covered the licensee's evaluation, pre-operational testing, and operation at operating plants, including the potential for adverse impact on plant SSCs and on plant programs. Some of these NRC inspections identified instances in which licensee's satisfied selected inspection requirements while others document instances in which licensees did not satisfy some of the inspection requirements.

The sample of NRC inspection reports reviewed by GIP representatives document NRC inspectors use of applicable inspection procedures for reviewing and assessing licensee performance for ISFSI, DCSS, and other heavy load campaigns for their potential impact on plant SSCs and programs. These inspection reports demonstrate NRC inspectors have used existing inspection procedures, as applicable, to assess licensee facilities and activities that involve: design basis, design change safety evaluations, maintenance and use of operating experience, and mitigative measures. They also specifically cover load drop analysis, control of documentation, single failure proof status, load path protection, and risk assessments of potential events and discovered conditions. The inspection reports document instances of findings for these areas as well as instances of no findings of significance. Accordingly, these inspection procedures and inspection reports demonstrate these items represent licensee compliance issues and as such, are not suitable for further assessment under the GIP.

Inspection procedures addressing cranes that lift spent fuel casks apply to program areas covered under inspection manual chapter (IMC) 2690 [i.e., 10 CFR Part 72 licenses (Office of Nuclear Material Safety and Safeguards (NMSS))] and under IMC 2515 [i.e., 10 CFR Part 50 licenses (Office of Nuclear Reactor Regulation (NRR))].

1. NRC Inspection Manual Chapter 2690, "Inspection Program for Dry Storage of Spent Reactor Fuel at Independent Spent Fuel Storage Installations and for Part 71 Transportation Packagings."

2. NRC Inspection Manual Chapter 2515, "Light Water Reactor Inspection Program – Operations Phase."

The Inspection Procedures (IP) identified below apply to various aspects of spent fuel storage casks, the cranes used to lift these casks, and the potential impact of spent fuel casks on operating plant structures, systems, and components, as indicated.

Review of 10 CFR 72.212(b) Evaluations (**IP 60856**, issue date 04/02/2003) and (**IP 60856.1** at Operating Plants, issue date 02/02/2004), apply to IMC 2690 and IMC 2515. These procedures are only used for general ISFSI licenses. These procedures require review of ISFSI activities for determination of no adverse impact on site operations or technical specifications documented in licensee evaluations of these activities. These procedures also require review of programs impacted by ISFSI operation (i.e., Emergency plan, Quality assurance program, Radiation protection program, and Training program) to verify that the licensee reviewed these programs to determine potential impact on their effectiveness and implement approved changes, as necessary. The general guidance directs inspectors to contact the SFPO project manager for obtaining NRR assistance (if needed) for questions involving how ISFSI activities potentially impact safety-related reactor SSCs. Specific guidance calls for inspectors to review the licensee's evaluation of ISFSI for any potential impact on the site or technical specifications including:

a. Impacts that could require 10 CFR 50.59 evaluations (and verify their proper performance).
b. Verify dropping a storage or fuel transfer cask inside the fuel handling or reactor building will not challenge any safety-related facility SSCs.
c. Verify storage and transfer cask pathways were approved in accordance with the licensee's heavy-load administrative requirements.
d. Verify the licensee's analysis verifies the weight of the DCSS is bounded by the UFSAR dropped cask analysis and meets NUREG-0612 requirements for controlling heavy loads, or fully analyzes the impact of a dropped DCSS on the reactor facility.
e. Verify licensee assumptions in the cask drop analysis are consistent with the NRC case SER.
f. Verify the fuel handling or reactor building cranes can safely move the storage of fuel transfer casks (reference NUREG 0612), including proper evaluation of any changes to interlocks or mechanical stops.
g. Verifications the licensee has adequately evaluated the transport pathways for loads, underground SSCs, interferences, temporary suspensions while transporting, right of way, roadway wear, security, traffic, fire hazards, etc.

3. Pre-operational Testing of Independent Spent Fuel Storage Facility Installations **(IP 60854**, issue date 03/28/2002) and (**IP 60854.1** at Operating Plants, issue date 09/05/2006) apply to IMC 2690 and IMC 2515. These procedures are used for general ISFSI licenses and for a specific licensed ISFSI. These procedures require verification that pre-operational test procedures for the DCSS loading, unloading, and transfer activities and their acceptance criteria meet the commitments and requirements specified in the DCSS safety analysis report (SAR), safety evaluation report (SER), certificate of compliance (CoC), 10 CFR Part 72, the site specific license and technical specifications as applicable, any related 10 CFR 50.59 and 72.48 evaluations, and 10 CFR 72.212(b) evaluations for general licensed ISFSIs. These procedures also require verification that pre-operational test procedures have been prepared, reviewed, approved, and independently verified and that they have sufficient overlap for critical activities including activities involved in transferring spent fuel from the spent fuel pool (SFP) to the ISFSI and for activities involved in retrieving spent fuel from a loaded DCSS in the ISFSI and returning it to the SFP. In addition, these procedures require verification that equipment used during pre-operational test activities has been tested

and / or evaluated for its impact on plant SSCs before performance of the pre-operational test. Finally, these procedures require inspectors to determine that responsibilities for specific activities relating to the ISFSI (i.e., design, component fabrication, construction, pre-operational testing, operations, maintenance, and surveillance testing) have been defined and the licensee has integrated responsibilities for these activities into the appropriate plant programs, including the Program for Control of Heavy Loads. This includes verification that these procedures fulfill the commitments and requirements specified in the SAR, SER, CoC, 10 CFR Part 72, the site-specific license and TS as applicable, any related 10 CFR 50.59 and 72.48 evaluations, and 10 CRF 72.212(b) evaluations for general licensed ISFSIs. These procedures provide substantial specific guidance that includes:

a. A. Review of pre-operational test procedures for DCSS loading, unloading, and transfer activities before conducting the on-site inspection.
b. Resolving different interpretations of construction code requirements.
c. Licensee guidance for time limits and contingencies for suspended loads.
d. Use of maximum weights including water.
e. Hold points for inspections.
f. Response to abnormal and emergency conditions and criteria for resumption of activities.
g. Capture of operational commitments from design bases documents and safety evaluations in the applicable procedures.
h. Pre-job briefings.
i. Testing and inspecting cranes and rigging and lifting equipment to verify that they can support the anticipated loads without compromising the licensing basis margins of safety and are compatible with the DCSS components.
j. Review of heavy load paths, crane single failure issues, and maximum DCSS weight versus crane capacity limits.

4. Operation of an Independent Spent Fuel Storage Installation (**IP 60855**, issue date 03/28/2002) and (**IP 60855.1** at Operating Plants, issue date 09/05/2006) apply to IMC 2690 and IMC 2515. These procedures require review of the plant specific license and TS for the DCSS being used, including relevant bulletins, Information Notices, or 10 CFR Part 21 reports issued related to ISFSI activities or the specific cask design. These procedures also require review of changes made to the programs and procedures involved in or potentially affected by ISFSI activities, including control of heavy loads, to verify that changes made were consistent with the license or CoC and did not reduce the effectiveness of the program. This review is performed to verify that these procedures still fulfill the commitments and requirements specified in the SAR, SER, CoC, 10 CFR Part 72, the site specific license and TS as applicable, any related 10 CFR 50.59 and 72.48 evaluations, and 10 CFR 72.212(b) evaluations. These procedures require verification that the licensee has performed either loading or unloading, as applicable, in a safe manner and in compliance with approved procedures. These procedures also require the inspector evaluation the effectiveness of the licensee's management oversight and QA assessments of ISFSI activities, for loading, unloading, or normal operations, as applicable. The general guidance notes that activities inside the reactor or fuel buildings (e.g., lifting of heavy loads or movement of spent fuel) may have a direct impact on safety-related SSCs. Therefore, activities potentially affecting safety-related SSCs should receive additional attention. This guidance tells inspectors to direct their questions on ISFSI activities affecting safety-related SSCs to the NRR project manager and that assistance may be obtained from SFPO (NMSS/SFPO) and

NRR. The specific guidance provides significant details including that inspectors review any changes to the crane, yoke, heavy loads program since the last inspection. If any changes were made, review the 10 CFR 50.59 or 72.48 safety evaluation that was performed. This specific guidance also refers the inspector to applicable MC 2515 Inspection Procedures, as appropriate for supplemental guidance in specific program areas, as applicable.

5. On-Site Fabrication of Components and Construction of an ISFSI (**IP 60853**, issue date 01/11/2004) applies to IMC 2690 and 2515. Inspectors use this procedure to determine whether ISFSI DCSS components are fabricated in accordance with the SAR, QA program, SER, CoC or site-specific license and technical specifications, and 10 CFR Part 72. It is also used to determine whether ISFSI construction activities are conducted in accordance with the QA program and to determine whether the licensee has reviewed the ISFSI activities for determination of no adverse impact on site operations and technical specifications. The IP's inspection requirements address specifications and 10 CFR Part 21, fabrication per design commitments including modifications and potential impact on plant SSC (i.e., relevant 10 CFR 50.59 or 72.48 evaluations), and QA activities for onsite fabrication as well as other requirements. General guidance addresses determining safety classification of SSCs, design changes, and component functionality for meeting design requirements. Specific guidance is provided for items including specifications, design and design changes, fabrication and examination techniques, design basis and design reviews, and associated QA activities.

The sample of 21 inspection reports summarized below document NRC inspectors use of the above inspection procedures, and others as noted in these inspection reports, for reviewing and assessing licensee performance for ISFSI, DCSS, and their potential impact on plant SSCs and programs. These inspection reports also document some examples of NRC inspections involving other heavy loads using other inspection procedures, as appropriate. These inspection reports demonstrate NRC inspectors have used the inspection procedures identified above, and others as conditions warrant, to assess licensee facilities and activities that involve design basis, design change safety evaluations, maintenance and use of operating experience, and mitigative measures. These inspection reports more specifically show NRC inspectors use existing inspection procedures to examine load drop analysis, control of documentation, single failure proof status, load path protection, and risk assessments of potential events and discovered conditions. These 21 inspection reports are listed in reverse chronological order and the applicable IP is shown in bold, except for cases where no IP was identified.

(1) POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORTS 05000266/2006004; 05000301/2006004; 05000266/2006009; 05000301/2006009, Dated July 20, 2006. This report documents the inspectors review of the licensees Operation of an Independent Spent Fuel Storage Installation (ISFSI) **(IP 60855.1)**. The inspectors reviewed the licensee's calculation of the capability of the auxiliary building to support a single failure-proof-crane with a 125-ton load during a seismic event. The inspectors identified a Severity Level IV, Non-Cited Violation of very low safety significance for the failure to perform a written evaluation in accordance with 10 CFR 50.59(d)(1) for calculation PBNP-305336-S01 to analyze the capability of the auxiliary building to support a single-failure-proof crane with a 125-ton load during a seismic event. A previous structural analysis qualified the auxiliary building to hold a 101-ton lifted load. In 2004, during an NRC inspection of the licensee's activities to demonstrate readiness to load fuel for dry storage, the inspectors determined that the license had not performed a structural analysis of the auxiliary building to show its capability to hold the NUHOMS casks during a seismic event. The licensee had previously concluded that

the auxiliary building could hold a maximum of 101-ton load during a seismic event. The previous analysis addressed loads that were imposed on the crane and the auxiliary building due to the VSC-24 cask design that was used prior to 2004. Subsequently, the licensee committed to perform a complete analysis of the auxiliary building and the crane to demonstrate the structural adequacy of the auxiliary building to lift the new casks prior to the next loading campaign. NRC review of this analysis was tracked as a URI in IR 07200005/2004-003. The licensee's next loading campaign was scheduled to take place in April 2006. The licensee completed the analysis before April 2006 and concluded that the auxiliary building could hold the crane and a maximum load of 125 tons during an earthquake. In reviewing this analysis, the inspectors determined that the licensee had not performed a 10 CFR 50.59 evaluation to determine whether a license amendment was required due to the increased design loads imposed on the crane and the auxiliary building due to the use of the NUHOMS casks. Subsequently, this issue was entered into the licensee's corrective action program, and a 50.59 screening (No. 2006-0063) and a full 50.59 evaluation (No. 2006-006) were performed. The licensee concluded in its evaluation that a license amendment was not required for the increase in the design load.

The inspectors determined that the licensee's failure to use the 10 CFR 50.59 process to evaluate the change of increasing the load on the auxiliary building is a licensee performance deficiency. Because issues involving 10 CFR 50.59 potentially affect the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. In accordance with the NRC Enforcement Policy, this finding is determined to be more than minor because there was a reasonable likelihood that the change requiring the 10 CFR 50.59 evaluation would require NRC review and approval prior to implementation. This finding has been reviewed by NRC management and is determined to be a Green finding, of very low safety significance. Because violations of 10 CFR 50.59 affect the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. In accordance with the NRC Enforcement Policy, this finding is determined to be more than minor because there was a reasonable likelihood that the change requiring the 10 CFR 50.59 evaluation would require NRC review and approval prior to implementation. This finding has been reviewed by NRC management and is determined to be a Green finding, of very low safety significance.

(2) Pilgrim Nuclear Power Station – NRC Integrated Inspection Report 05000293/2006003, Dated April 1, 2006. This report documents the inspectors review of Maintenance Risk Assessments and Emergent Work Control **(IP 71111.13)**, including the risk associated with the control of heavy loads while handling a CNS 3-55 waste shipment cask in the Reactor Building. The inspectors identified a Severity Level IV Non-Cited Violation associated with the failure to perform an adequate safety evaluation per 10 CR 50.59. Contrary to 10 CFR 50.59, a screening safety evaluation for handling of a 35 ton cask in the spent fuel did not provide an adequate basis to demonstrate that the evaluation for use of a heavier cask did not change the evaluation methods approved by the NRC staff in 1985 for the control of heavy loads per NUREG 0612 commitments, as described in the UFSAR and the Pilgrim licensing basis. The finding was determined to be more than minor because the inspectors could not reasonably determine that the proposed cask handling activity would not have required NRC approval without subsequent significant enhancements to the 50.59 safety evaluation. The conditions associated with the finding (i.e., the potential drop of a loaded cask) were determined to be of very low safety significance because they did not result in the loss of operability of a safety system. Because the issue affected the NRC's ability to perform its regulatory function, this finding was dispositioned using the traditional enforcement process and was classified at Severity level IV because the violation of 10 CFR 50.59 involved conditions evaluated as having

very low safety significance by the SDP.

(3) PALISADES NUCLEAR PLANT NRC INSPECTION REPORT 05000255/2005012, Dated January 25, 2006. This report documents the inspectors review of Operator Performance During Non-routine Evolutions and Events **(IP 71111.14)** and Operation of an Independent Spent Fuel Storage Installation (ISFSI) **(IP 60855.1)**. The inspectors reviewed the loading procedures and observed activities associated with the loading and transfer of two NUHOMS 32 PT casks. On October 11, 2005, while raising a dry fuel storage (DFS) cask from the spent fuel pool following loading of the cask, the emergency brake on the crane engaged. The engaged emergency brake stopped movement of the load resulting in suspension of the load partially out of the pool. This was a lift of the first loaded canister and transfer cask out of the Spent Fuel Pool (SFP). The top of the suspended cask was approximately 5 feet above the water surface. The inspectors observed and evaluated the license's response during the event. The inspectors also reviewed the crane's annual inspection records, work orders associated with trouble shooting activities, and the Root Cause Analysis Report associated with this event. The inspectors evaluated the adequacy of the short and long-term corrective actions that the licensee proposed and initiated to prevent future occurrences of similar issues. For this non-routine event, the inspectors reviewed operator logs, plant computer data, and strip charts as appropriate to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures. The inspectors observed licensee activities to determine the cause of the brake's engagement and to lower safely the load back into the pool. The inspectors identified one finding of very low safety significance and an associated non-cited violation when plant personnel performed activities outside the scope of the work package used to inspect the spent fuel pool crane. During troubleshooting activities, the workers exceeded the bounds of the approved work package by manipulating the brake release. This finding represented a violation of the license by performing work contrary to requirements specified by NUREG-0612. Corrective actions included reinforcing site standards for procedural adherence as well as successfully lowering the DFS cask. The licensee entered the item in the Corrective Action Program. The finding was not suitable for evaluation under the SDP. However, because the actions by the worker did not result in any load motion and both crane brakes remained set, NRC management determined the finding to be of very low safety significance (Green).

(4) Sequoyah Nuclear Power Plant – NRC Integrated Inspection Report 05000327/2005005 and 05000328/2005005, Dated January 24, 2006. This report documents the inspector's Review of the Operation of an ISFSI **(IP 60855.1)**. The inspectors reviewed the ISFSI to verify that normal operations were conducted in a safe manner in accordance with approved procedures and without undue risk to the health and safety of the public. The inspectors walked down the ISFSI pad to assess the material condition of the casks. The inspectors also reviewed dry cask storage records from the previous loading campaign to verify that the licensee had identified each fuel assembly placed in the ISFSI, identified the parameters and characteristics of each fuel assembly, maintained controlled copy records of all spent fuel placed in the ISFSI, maintained duplicate records at a separate location, and performed a physical inventory every 12 months. Documents reviewed are listed in the attachment.

The inspectors identified no findings of significance from this inspection.

This inspection report also closed Unresolved Item (URI) 05000327/2004009-03, Inadequate Design Control for Calculation 44N300C7, 125-Ton Crane-Auxiliary Building (ISFSI). This issue was thoroughly addressed in Inspection Report 07200034/2005002, issued on January 9, 2006. The URI was closed based on the findings of that inspection.

(5) Sequoyah Nuclear Power Plant – Independent Spent Fuel Storage Installation (ISFSI) Dry Run and Fuel Loading NRC Inspection Report 07200034/2004001, Dated September 10, 2004. This report documents the NRC's August 11, 2004, inspection of Sequoyah Nuclear Power Plant Independent Spent Fuel Storage Installation (ISFSI) using **IP 60853, IP 60854, IP 60854.1, and IP 60855**. The inspection report documents the inspection of activities conducted under the ISFSI license as they relate to safety and compliance with the Commission's rules and regulations. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. These inspection included observation of activities associated with pre-operational testing program and the loading of your first ISFSI cask. The inspections were conducted to confirm compliance of your program and activities with the requirements specified in the license, technical specifications, Final Safety Analysis Report and the NRC's Safety Evaluation Report for the Holtec cask system. The enclosed report presents the results of this inspection. The NRC conducted onsite inspections of the activities associated with the licensee's cask storage program. The NRC observed licensee activities including loading, drying, helium backfilling and welding activities of the first cask. NRC Inspectors were also present for the heavy lift of the loaded canister, movement of the canister, and the lowering of the canister into the concrete cask. The NRC inspections focused on the licensee's efforts to demonstrate that adequate equipment, procedures, and personnel were in-place to safely move spent fuel from the reactor spent fuel pools to the ISFSI pad. The pre-operational test requirements covered all key activities related to loading a cask and moving the cask to the ISFSI pad. Throughout the demonstrations observed by the NRC, the Sequoyah staff functioned professionally and performed their assigned functions safely. The initial cask loading was completed successfully following an interruption due to problems associated with the Auxiliary Building crane. Two NRC identified non-cited violations (NCVs) were identified during the inspection. One NCV involved material storage deficiencies and the other NCV involved inadequate corrective actions related to the Auxiliary Building crane that was used to lift the cask. One unresolved item (URI) for crane design issues was also identified by the NRC.

During this inspection, the inspectors reviewed the Auxiliary Building crane weld crack repairs and modification that were performed by the licensee as a result of problems encountered during initial cask loading. The inspectors also observed the 100% and 125% crane load tests following repairs, including independent inspection to check any weld or base metal cracks or deformations. The inspectors reviewed crane history, PERs, calculations, evaluation, Work Orders (WOs), welder qualification and certification, crane operator and flager qualification and certification, root cause analysis and evaluation, and other documents. This inspection resulted in One violation and one Unresolved Item (URI) were identified. The violation involved inadequate corrective actions and inspections resulting in failure to identify problems in the critical welds in the Auxiliary Building crane. The URI involved the adequacy of design control for calculation 44N300C7, 125 Ton Crane-Auxiliary Building. Additional details are provided in the inspection report (ADAMS ML042740167).

(6) Arkansas Nuclear One – NRC Integrated Inspection Report 05000313/2003005, 05000368/2003005, and 07200013/2003003, Dated January 29, 2004. This report documents the inspectors review of the licensee's Pre-operational Testing of an Independent Spent Fuel Storage Installation **(IP 60854)** and Review of 10 CFR 72.212(b) Evaluations **(IP 60856)**. The scope of this inspection included the following.

An NRC inspection of selected pre-operational testing activities related to use of the Holtec HI-STORM 100 cask system at the ANO independent spent fuel storage installation (ISFSI) was conducted on November 18-20, 2003. ANO had completed loading 24 Sierra Nuclear Casks VSC-24s preparing for the next loading campaign using the Holtec HI-STORM 100 cask

system. The first Holtec cask was scheduled for loading during December 2003.

License Conditions 10.a through 10.j of the Holtec Certificate of Compliance 1014 required pre-operational testing prior to the first use of the HI-STORM 100 cask system. ANO completed all the requirements in License Condition 10 by demonstrating the use of the Holtec systems during NRC inspections on March 17-20 and April 21-24, 2003, and this inspection and by demonstrations observed by the NRC during the pre-operational inspections on the Sierra Nuclear casks. Inspection Report 05000313/1996016; 05000368/1996016; 07200013/1996001 dated July 31, 1996, and Inspection Report 05000313/2003003; 05000368/2003003; 07200013/2003002 dated September 8, 2003, provide documentation of the previously demonstrated activities. The following list provides a summary of the completed demonstrations as required by Licensee Condition 10.

10.a Moving the canister and transfer cask into the spent fuel pool (observed in NRC Inspection Report 07200013/1996001).

10.b Preparation of cask for fuel loading (observed in NRC Inspection Report 07200013/1996001).

10.c Selection and verification of specific fuel assemblies to ensure type conformance (Document 1302.028, "Fuel Selection Criteria for Dry Storage," was reviewed during this inspection. This document provided a description of the fuel selection criteria being used by ANO and was consistent with the Technical Specification requirements specified in the Holtec Certificate of Compliance 72-1014).

10.d Loading specific assemblies into the canister, using a dummy fuel assembly, including independent verification (observed in NRC Inspection Report 07200013/1996001).

10.e Remote installation of the canister lid and removal of the canister and transfer cask from the spent fuel pool (observed in NRC Inspection Report 07200013/1996001).

10.f Canister welding, nondestructive examinations, hydrostatic testing, draining, moisture removal by vacuum drying or forced helium dehydration, helium backfilling, and leakage testing (observed in NRC Inspection Report 07200013/2003002).

10.g Transfer cask upending and downending. (Not applicable to ANO. All cask handling operations are performed in the vertical position with no upending or downending operations.)

10.h Transfer of the canister from the transfer cask to the overpack (observed during this inspection).

10.I Placement of a cask at the ISFSI (observed during this inspection).

10.j Cask unloading, including cooling fuel assemblies, flooding canister cavity and removing canister lid welds. (Cask unloading was observed during this inspection. All other activities were observed in NRC Inspection Reports 07200013/2003002 and 07200013/1996001.)

ANO's Pre-operational demonstrations were performed using a work plan that had been written, tested, revised, and approved to support the evolution. All personnel knew their duties and carried them out efficiently. The activities observed during this inspection complete the NRC's Pre-operational inspection program requirements for the use of the Holtec HI-STORM 100 cask

system at ANO.

During the 125 percent load test conducted on the new Ederer crane in January 2003, the NRC had identified a discrepancy between the test acceptance criteria in the ANO procedures versus the acceptance criteria specified by Ederer, the crane manufacturer. This issue was documented as an inspector followup item (IFI) in NRC Inspection Report 05000313/2003009; 05000368/2003009; 07200013/2003001 dated February 7, 2003. The licensee also opened CR ANO-1-2003-00364 concerning the issue. On June 2, 2003, Ederer issued a letter to ANO stating that the 125 percent load test acceptance criteria used by ANO met the ASME B30.2 1997 criteria and was acceptable to Ederer. This closes the open item (IFI 07200013/2003001-01).

The 10 CFR 72.212(b) evaluation for the use of the Holtec casks at the ANO site was reviewed. The evaluation encompassed the environmental conditions at ANO and identified the licensee's programs that were revised to incorporate the use the Holtec HI-STORM 100 cask system at ANO. No discrepancies or issues were identified.

The inspectors identified no finding from this inspection.

(7) North Anna Power Station – NRC Integrated Inspection Report No. 05000338/2003005, 05000339/2003005 And Independent Spent Fuel Storage Installation Report No 07200016/2003002, Dated January 26, 2004. This report documents the inspectors observation of dry cask loading **(IP 60855)**, which included: observed loading spent fuel assemblies into the spent fuel dry storage cask; verified positive engagement of lifting devices being positioned; lifting of the loaded cask above the water surface; moving the loaded cask to the cask setting area by following the heavy load lifting path; confirmation that activities confirmed to licensee's procedures; review of training certificates and qualification records for crane operators and cask loading operators; and review of spent fuel cask crane periodic inspection, functional test, and maintenance records.

The inspectors identified no findings from this inspection.

(8) Oconee Nuclear Station – Integrated Inspection Report 05000269/2003005, 05000270/2003005, and 05000287/2003005, and Independent Spent Fuel Storage Installation Inspection Report 72-04/2003001, Dated January 26, 2004. This report documents the inspectors review of the licensee's operation of an ISFSI **(IP 60855)**, which included: the licensee's procedure for loading spent fuel shipments to the ISFSI and associated Problem Investigation Process Reports to verify that the ISFSI shipment activities for 2003 were performed in a safe manner and in compliance with the approved procedure.

The inspectors identified no finding from this inspection.

Related to the broader topic of crane safety, this inspection report also documents the inspectors review of the licensee's steam generator replacement program, lifting, and transportation **(IP 50001)**. The scope of this inspection included the following. The inspectors reviewed the adequacy of the SGRP rigging and handling program as described in ON-13086 AS6, "Steam Generator Rigging and Handling," Rev. 0E1 to verify compliance with regulatory requirements, appropriate industrial codes, and standards, ANSI N45.2.15, Generic Letter 81-07 and NUREG 0612.

The inspectors examined portions of the SGRP lifting equipment necessary to perform steam

generator rigging and transport, design evaluation/erection/use of the Outside Lift System (OLS) and Temporary Lifting Device (TLD), Hatch Transfer System (HTS), and a Self Propelled Modular Transport (SPMT). The inspectors reviewed the applicable engineering design, modification and analysis associated with SG lifting and rigging including: crane and rigging equipment, steam generator drop analysis, safe load paths, and load drop protection. The inspectors determined if appropriate load tests and functional tests were performed or documented in accordance with the ASME/ANSI code for both the TLD, OLS, and lifting links. The inspectors determined if the TLD and OLS cranes were operated by qualified and certified personnel, and that wire ropes and synthetic slings used during heavy lifts were appropriately tested and inspected prior to use. The inspectors determined if the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

For changes to the facility design as described in the UFSAR, the inspectors reviewed the 10CFR 50.59 screens for modification packages. For those modification packages that did not involve a change to the facility as described in the UFSAR, a 10CFR 50.65 Risk Assessment was done and reviewed by the inspectors. The inspectors determined if Oconee Operations was aware of the potentially impacted Instrument Air (IA) and Low Pressure Service Water (LPSW) systems and if they had a contingency plan to isolate those two systems in case of an accident.

The inspectors also observed various portions of the original steam generators (OSG) being lifted from the steam generator cubicle through the temporary opening in the reactor building utilizing the TLD installed in the Containment Building, the HTS, the OLS, onto the SPMT. The inspectors also observed various portions of the replacement steam generators (RSG) being lifted back into the containment. During these observations the inspectors performed visual inspections of the TLD, HTS, OLS, and SPMT. For the task of rigging and movement of the SGs, the inspectors reviewed the work packages and procedures for content, technical adequacy and to verify that appropriate line items had been signed off and that required pre-lift equipment inspections had been performed and documented in the enclosures provided. This review was also to verify that operating experience was utilized and reflected in the procedures. Documents reviewed during this inspection are listed in the Attachment to this report.

The inspectors identified no finding from this inspection.

(9) NRC Inspection Report 50-206/03-10; 50-361/03-10; 50-362/03-10; 72-41/03-01 For The Pre-Operational Testing And First Canister Loading At The Independent Spent Fuel Storage Installation, Dated January 6, 2004. This report documents NRC inspectors review of the San Onofre Nuclear Generating Station (SONGS) **pre-operational testing of an ISFSI (IP 60854), operations of an ISFSI (IP 60855), licensee evaluations (IP 60856 and IP 60857), and ISFSI security (IP 81001)**. The inspections conducted by the NRC of Southern California Edison's dry cask storage project provided a comprehensive evaluation of the licensee's compliance with the requirements in the NUHOMS Certificate of Compliance No. 72-1029, Technical Specifications, Final Safety Analysis Report, NRC's Safety Evaluation Report and 10 CFR Part 72. The inspection consisted of a team of eleven NRC inspectors performing inspections of various phases of activities over a period from June 26 through October 1, 2003. Sixteen technical areas were reviewed during the inspections including such topical areas as fuel verification, security, radiological programs, quality assurance, training, and heavy loads. During the inspections, the licensee conducted numerous demonstrations for NRC observance related to the operations of equipment and the implementation of procedures to verify that all required aspects of the use of the NUHOMS cask system at the SONGS site had been adequately incorporated into site programs and procedures. The licensee had integrated

together many of the programs for the Part 50 reactor operations and the Part 72 ISFSI operations to allow for an efficient use of site resources.

The scope of this inspection that involved the licensee's cranes handling spent fuel casks includes the following.

Heavy Loads
The licensee had incorporated the special requirements related to the ISFSI project into the site heavy loads programs and procedures. Crane operators interviewed during the ISFSI inspections were knowledgeable of the special handling requirements related to the heavy spent fuel casks.

Special lifting device height limits and temperature restrictions during movement of the casks had been incorporated into the licensee's procedures consistent with the requirements in the Certificate of Compliance. This included the requirement to perform an inspection of the cask for damage after any drop of 15" or higher.

The maximum weight of the loaded transfer cask had been determined by the licensee to verify the adequacy of the lift capability of the cranes planned for use in moving the spent fuel. A loaded cask being removed from the spent fuel pool represented the maximum weight to be lifted and exceeded the lifting capacity of the current Unit 1 crane. The licensee had purchased new single failure proof cranes for installation and use during the cask loading operations. The Unit 3 crane was rated at 125 tons. The Unit 1 crane would be up-rated from 100 tons to 105 tons to account for the estimated 104.1 ton weight of the loaded cask.

A safe loads path had been identified and analyzed for moving the spent fuel from the spent fuel pool to the ISFSI. Provisions were established in procedures and through the use of limit switches to prevent the crane from moving the loaded cask outside the boundaries of the safe load path while in the fuel building. Calculations were performed for the roadway between the plant and the ISFSI to verify that the path was structurally capable of handling the weight of the loaded transfer cask and trailer.

The licensee's heavy loads procedural requirements related to the transfer cask trunnions and the slings used for lifting the canister lid were verified against industry standards and manufacturer requirements for load tests, safety margins and inspection/maintenance.

The adequacy of the transport trailer for the expected weight of a loaded canister and the ability of the transport trailer to safely secure and move the cask to the ISFSI was verified.

Pre-Operational Testing

The licensee successfully completed all the required pre-operational test requirements specified by the Certificate of Compliance. This include the loading, welding, drying, and backfilling of a canister and the unloading of a sealed canister. A weighted canister was used to demonstrate heavy load activities, transport between the Unit 3 facility and the ISFSI and insertion/removal of a canister into the concrete storage module.

Procedures and Technical Specification Requirements

The licensee had developed the required operating procedures and programs required by the Certificate of Compliance. The adequacy of the level of detail in the procedures was verified by

reviewing requirements in the Final Safety Analysis Report related to handling the cask at the spent fuel pool. All requirements reviewed had been incorporated as operational steps in the licensee's procedures.

Quality Assurance

The licensee had implemented their approved reactor facility Part 50 quality assurance program for the activities associated with the ISFSI. Effective implementation of the program was observed for all phases of ISFSI activities including procurement, control of measuring equipment, corrective actions, design control, receipt inspections, storage and audits.

Safety Reviews

Changes to the site related to the construction and operation of the ISFSI were being evaluated in accordance with 10 CFR 72.48 and 10 CFR 50.59 requirements. No issues were identified during the review of selected safety screenings.

The inspectors identified no finding from this inspection.

(10) Arkansas Nuclear One – NRC Integrated Inspection Report 05000313/2003004 and 05000368/2003004, Dated November 3, 2003. This report documents closure of a finding from previous inspection report 0500313, 368/2003002-02: Failure to Obtain a License Amendment for Upgrade of the Spent Fuel Area Crane as described below. This inspection report documents inspectors review of Permanent Plant Modifications **(IP 71111.17A)**.

The inspectors identified a Severity Level IV violation of 10 CFR 50.59 when the licensee failed to initially submit a license amendment request for a modification to the L-3 spent fuel area crane. In 2001, the licensee changed vendors and styles of spent fuel storage casks. Part of this change required modification to upgrade the L-3 spent fuel area crane to lift the newer, heavier Holtec casks. For this modification, the licensee conducted a 10 CFR 50.59 evaluation and concluded that the proposed modification did not require a license amendment. The licensee concluded that the upgraded crane design met the requirements of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Therefore, the licensee concluded that the upgraded crane design was acceptable for implementation without the need for an NRC license amendment.

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

The inspectors, managers from the NRC Region IV office, and representatives of the NRC Office of Nuclear Reactor Regulation informed the licensee of this conclusion in a telephone call on February 13, 2003. The inspectors also informed the licensee that failure to submit a license amendment request for this modification was a potential violation of 10 CFR 50.59. The

licensee entered this issue into its corrective action program as CR ANO-C-2003-0092. The licensee subsequently submitted a license amendment request for the Crane L-3 modification to the NRC on February 24, 2003.

This is an item for traditional enforcement because it involves a violation of 10 CFR 50.59, an issue which impacts NRC oversight ability. The inspectors considered this issue more than minor because there was a reasonable likelihood that the change would require NRC review and approval prior to its implementation. In accordance with NRC enforcement procedures, the significance of this finding was evaluated using the SDP in order to assign a severity level. The finding was determined to affect the initiating events cornerstone objective attributable to fuel handling equipment performance. The finding was then found to not screen as risk significant due to a seismic, fire, flooding, or severe weather initiating event, and therefore was determined to be of very low safety significance. The inspectors also factored in their analysis the fact that the upgraded crane had not been used to transport a loaded spent fuel storage cask and was under administrative controls preventing its use in this manner, pending approval of the license amendment.

10 CFR 50.59 requires, in part, that a licensee obtain a license amendment implementing a proposed change to the facility as described in the final safety analysis report if the change would create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. Contrary to the above, prior to February 2003, the licensee did not submit a license amendment for a change to their facility, specifically the modification to the L-3 spent fuel area crane, when the change created the possibility for a malfunction of the L-3 crane that would result in a 130 ton load drop. The change to the final safety analysis report for the crane modification erroneously stated that the crane was immune to potential dropped loads, but the actual change to the facility created a possibility of the drop of a 130 ton cask, which exceeded the licensee's previously evaluated load drop analysis discussed in Section 15.1.23.1 of the Unit 2 final safety analysis report for a 100 ton load drop. The licensee subsequently submitted a license amendment request on February 24, 2003, and did not transport loaded spent fuel storage casks as detailed in CR ANO-C-2003-0092. Because this failure to submit a license amendment request is of very low safety significance and was documented in the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000313, 368/2003004-05). Because the SDP determined that this issue was of very low safety significance, it was categorized as Severity Level IV in accordance with the NRC Enforcement Policy.

(11) Sequoyah Nuclear Power Plant – NRC Integrated Inspection Report 05000327/2003005, 05000328/2003005, and 07200034/2003001, Dated October 27, 2003. This report documents NRC inspection that included Review of Quality Assurance and Plant Modification for Independent Spent Fuel Storage Installation that included the inspection scope of **IP 60851, IP 60853, and IP 60854**. The inspectors reviewed the quality assurance program, plant modification, and construction activities associated with the ISFSI construction project. The inspectors reviewed the TVA submittal to the NRC dated July 6, 1999, with respect to its intention to apply the previously approved 10 CFR 50, Appendix B, Quality Assurance Program to activities at the Sequoyah ISFSI, and self-assessment report SQN-PROJ-03-002, in order to determine the adequacy and effectiveness of recent and on-going ISFSI activities at Sequoyah and Corporate Nuclear Fuel. The inspectors reviewed Design Change Notices (DCNs), design calculations, records, an Auxiliary Building crane modification and load test, and the crane operator qualification and medical records in order to determine the adequacy and compliance with the procedures. The inspectors reviewed the safety reviews under 10 CFR 50.59 and

calculations to ensure that the Auxiliary Building floors were sufficient to support cask loads and transfer associated weight during the spent fuel cask operations. The inspectors reviewed the Auxiliary Building crane to ensure that it had been modified to a single-failure-proof crane and reviewed load tests in order to verify the crane load capacity for the spent fuel casks. The inspectors walked down the cask transport route, a concrete overpack construction pad, and the cask concrete storage pads, including fence and lighting, to verify completion. The inspectors measured the concrete pad size, the distances between the pads and the fence, and weld sizes and member sizes for the half of the work platform to be installed in the spent fuel pool cask pit area. The results of the measurements were compared to the design drawings. The inspectors also walked down the spent fuel pool area to review the Auxiliary Building crane modification and cask pit stands and portions of the work platform already installed at the pool pit area for the preparation of cask loading operation and compared results to the design specification, procedures, drawings, and Holtec HI-STORM 100 FSAR.

The inspectors reviewed corrective action including the violation response, revised procedure, and records of air content and concrete cylinder compressive load testings for Violation (VIO) 72-34/2002-001-01, Inadequate Procedure to Use the Correct Air Content Acceptance Criteria for Concrete to Ensure Adequacy. The inspectors also reviewed Problem Evaluation Reports (PERs) 02-015393-000 and 02-013982-000 associated with this issue. This item was considered closed based on the records reviewed.

The inspectors identified no findings of significance from this inspection.

(12) NRC Inspection Report 50-528/02-08; 50-530/02-08; 72-44/02-03, Dated June 9, 2003. This report documents NRC inspections, conducted between November 11, 2002, and March 21, 2003, at Palo Verde Nuclear reactor facility to evaluate the dry cask storage activities for the ISFSI. The inspections included observation of activities associated with the pre-operational testing program and the loading of the first cask to confirm program and activities comply with the requirements specified in the license, technical specifications, FSAR, and NRC SER for the cask system. The NRC conducted four onsite inspections of the activities associated with the licensee's pre-operational test program. The NRC inspectors observed the loading of the first cask and the heavy lift of the loaded canister from the cask loading pit, movement of the canister from the decontamination pit to the concrete cask, and the lowering of the canister into the concrete cask. The NRC inspections focused on the licensee's efforts to demonstrate that adequate equipment, procedures, and personnel were in place to safely move spent fuel from the reactor spent fuel pools to the ISFSI. The pre-operational test requirements covered all key activities related to loading a cask and moving the cask to the ISFSI. Demonstrations also included the process for unloading spent fuel from a cask, should that be necessary. The primary inspection procedures used for guidance during the pre-operational inspections were **IP 60854, IP 60855, IP 60856, IP 60857, and IP 81001**. The NRC inspectors reviewed 15 key technical areas against the requirements in the NAC-UMS FSAR, NAC-UMS CoC, the technical specifications associated with the NAC-UMS cask design, and 10 CFR Part 50, Part 72, and Part 73. The ISFSI activities specifically related to the licensee's use of the facility crane in handling spent fuel casks are summarized below.

The licensee had completed an evaluation of the Palo Verde reactor programs to verify compliance with the conditions of the NAC-UMS cask design, Certificate of Compliance, Final Safety Analysis Report and requirements in 10 CFR Part 72. The evaluation included a review of the Palo Verde Part 50 programs related to emergency planning, radiation protection, training, quality assurance and various other programs. Several exemptions were requested and were received from the NRC related to seismic criteria for the ISFSI pad, contamination

limits for the cask and record retention requirements (Attachment 2 - General License).

The licensee had installed a new trolley for the fuel building crane and had conducted an extensive analysis and review to determine that no single failure of the new systems could result in failure of the crane to maintain the load. The new system included a below -the hook lifting device called the SAFLIFT. The SAFLIFT replaced the yoke and slings that would normally be used for heavy load activities of moving the canister and transfer cask (Attachment 2 - Heavy Loads).

The licensee had established a safe loads path for moving the loaded canister such that it was not moved over the spent fuel pool. The cask would be near the safety related air handling system in the fuel building when the loaded transfer cask was placed on top of the concrete cask for insertion of the canister. However, the single failure proof aspects of the new fuel building crane allowed the cask to be within the “zone of influence” around the air handling system (Attachment 2 - Heavy Loads).

The licensee conducted an extensive pre-operational test program to prepare for the loading of the first cask. The NRC observed the required demonstrations during four inspection trips to the site. The first attempt to demonstrate welding of the canister lid resulted in a number of issues identified. As a result, Palo Verde conducted a second demonstration which resulted in a very high quality weld. Demonstrations related to heavy loads and the vacuum drying and helium backfill operations were very successfully performed. Personnel assigned to the ISFSI project were knowledgeable in their work assignments and the design aspects of the cask system and participated in the Pre-operational tests realistically as if an actual canister was being loaded (Attachment 2 - Pre-Operational Tests).

The licensee had incorporated the appropriate procedural information from Chapter 8 of the NAC-UMS Final Safety Analysis Report into the Palo Verde procedures for loading, sealing, moving and unloading a cask. Written procedures for all activities related to cask loading and ISFSI operations had been developed (Attachment 2 - Procedures & Tech Specs).

A considerable amount of time during this inspection was directed toward review of safety evaluations associated with the ISFSI and in particular, the safety reviews associated with the replacement of the fuel building crane trolley. All safety reviews and screenings completed by the licensee were well documented with a good level of detail (Attachment 2 - Safety Reviews).

The training program for personnel assigned to the ISFSI provided a good basis for understanding the requirements and safe practices associated with dry cask loading operations (Attachment 2 - Training).

The inspectors identified no finding from this inspection.

(13) This North Anna Power Station – NRC Integrated Inspection Report nos. 50-338/03-02 and 50-339/03-02, Dated May 5, 2003, relates to the broader topic of crane safety. This inspection report documents NRC inspectors review of the reactor pressure vessel head replacement lifting and transportation program activities in accordance with **IP 71007**. The inspectors reviewed the licensee’s heavy load lifting and transportation program to ensure that it met the Updated Final Safety Analysis Report (UFSAR) and regulatory requirements for application to the Unit 1 reactor head replacement. The inspectors walked down the containment to review the polar crane condition and the head runway installation as part of preparation to remove the old head out and lift the new head into containment. The inspectors also walked down the

outside at the containment wall opening area to review the outside runway and its supporting structure and the 300 Ton Manitowoc-M250 Mobile Crane condition. The inspectors observed the old RPVH lifted above the refueling floor from storage area, set on the cart, pushed out to the outside platform from the containment using the runway installed, and lifted on the top of a transporter.

The inspectors reviewed the procedures, polar crane maintenance and inspection records, crane operator qualification records, Work Plan and Inspection Record (WPIR), drawings for head lifting steps, runways, and supporting structures, crane lifting capacity, mobile crane M-250 inspection and certificate, mobile crane stability evaluation, mobile crane load drop analysis, the structural calculation for the platform and supporting structure, and design change packages.

The inspectors identified no finding from this inspection.

(14) Dresden Nuclear Power Station NRC Inspection Report 50-237/03092; 50-249/03-02, Dated April 30, 2003. This inspection report documents resolution of Unit 2/3 Crane Issues **(IP 60855)** involving unresolved items from Inspection Report 0720037/2001-002. The resolution is summarized as follows (the inspection report provides additional details).

In order to resolve the licensing basis of the Reactor Building (RB) Superstructure and the crane bridge and trolley, the inspectors requested technical assistance from the Office of Nuclear Reactor Regulation (NRR). Task Interface Agreement (TIA) 2001-13 dated September 28, 2001, was issued to request review and comment on a backfit analysis related to the long term use of the Unit 2/3 RB crane to lift heavy loads at the Dresden Nuclear Power Station. In order to respond to questions related to the original licensing basis of the crane, the licensee issued April 12, 2002, and July 8, 2002, responses to an NRR request for additional information.

The licensee's response referenced a new revision, Revision 1, to calculation DRE 98-0020, which analyzed the RB superstructure with the crane loaded for forces imposed from both the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE).

On February 2, 2003, NRR issued a response to TIA 2001-13, in which NRR concluded, based on the new information and calculations provided by the licensee, that compliance with the licensing basis for the RB crane established in 1976 will provide an acceptable level of safety. The NRC determined that no further backfit analysis was necessary. This addresses previously identified compliance issues from inspection report 07200037/2001-002.

The inspectors initially identified these issues during an inspection of the Dresden dry cask storage and handling facilities. However, the safety impact of the findings regarding the Unit 2/3 crane system were primarily related to the operation of Dresden Units 2 and 3. Therefore, the Unresolved Items initially opened related to 10 CFR Part 72 were closed using 10 CFR Part 50 criteria.

The inspectors determined that these findings were greater than minor in accordance with Inspection Manual Chapter [IMC] 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated February 2, 2003. The findings dealt with Mitigating Systems and Barrier Integrity Cornerstone objectives related to the attributes of design control and equipment performance. Assumptions regarding the risk significance of Dresden RB crane issues were coordinated with a Region III Senior Reactor Analyst. Bounding assumptions

included a potential rapid lowering of the heavy load while over the Unit 2 suppression pool due to heavy load handling system failures. Due to the low seismic initiating event frequency, the short duration of time that the heavy loads were suspended on the RB crane, the nature of the load path and load lift controls, and the recent licensee calculations which demonstrated that the RB superstructure will support the crane with lifted load in a seismic event, the findings were determined to be of very low safety significance (Green).

(15) NRC Inspection Report 50-313/03-09; 50-368/03-09; 72-13/03-01, Dated February 7, 2003. This inspection report documents the NRC's completion, on January 14-15, 2003, of the routine annual inspection of the ISFSI at Arkansas Nuclear One, Units 1 and 2 facilities and observations of crane tests for the new fuel building crane. The inspection involved use of **IP 60855, IP 60857, and IP 60854**. The inspection focused on ANO's compliance with the regulatory requirements and license commitments associated with the ISFSI and with the onsite qualification program to certify the crane as a single failure proof crane.

The inspectors identified no finding from this inspection.

(16) NRC Inspection Report 50-397/02-08; 72-35/02-01, Dated November 1, 2002. This report provides the results of the Nuclear Regulatory Commission's (NRC) team inspections conducted between June 4 and September 20, 2002, at your Columbia Generating Station nuclear reactor facility to evaluate the dry cask storage activities related to your newly constructed Independent Spent Fuel Storage Installation (ISFSI) and to observe the loading of your first cask. The inspections were conducted to confirm compliance of your program and activities with the requirements specified in the certificate of compliance, technical specifications, and Final Safety Analysis Report (FSAR) for the Holtec HI-STORM 100S cask system (Certificate No. 1014) being used at your ISFSI. The inspectors used the following inspection procedures during this inspection: **IP 60801, IP 60851, IP 60854, IP 60855, IP 60856, IP 60857, IP 81001, and IP 83750**.

The NRC conducted an extensive evaluation of the licensee's program for the safe handling and storage of spent fuel at their ISFSI, observed the pre-operational test demonstrations, and observed the loading of the first cask. This inspection effort consisted of an in-depth evaluation of the licensee's programs, procedures, training and staff qualifications against the requirements in 10 CFR Part 72, the HI-STORM 100S Certificate of Compliance and the Final Safety Analysis Report (FSAR). The pre-operational testing program required the licensee to demonstrate, through the use of actual equipment and mock-ups, that preparations had been completed to safely load a cask with spent fuel and move the cask to the ISFSI. These required demonstrations were specified in Condition 10 of the Certificate of Compliance. Condition 10 required pre-operational testing of the loading, closure, handling, unloading, and transfer of the HI-STORM 100S cask system to be conducted by the licensee prior to the first use of the system to load spent fuel assemblies.

Columbia Generating Station developed a pre-operational test plan which consisted of six exercises encompassing the required demonstrations and developed a schedule for conducting the exercises under observation of the NRC. The exercises were conducted between June and September 2002. Five trips to the Columbia Generating Station site were completed by the NRC. Inspectors from the NRC Region IV office, NRC headquarters and the NRC's field office for the Yucca Mountain Project participated in the inspections. The inspections were a comprehensive review of the activities associated with safely loading a cask, placing the cask into the ISFSI and maintaining an ongoing program to ensure the cask will be safely stored. In addition, programs being implemented by the reactor facility under their Part 50 license that

would be used to support the cask loading and storage activities were reviewed to ensure adequate integration between the two programs.

The scope of this inspection that involved the licensee's cranes handling spent fuel casks includes the following.

Pre-Operational Test Program

• The licensee was required by the certificate of compliance to conduct a pre-operational test program to demonstrate readiness to load spent fuel. The NRC conducted five inspections over a 4-month period to observe the required demonstrations. All required activities were successfully completed and the licensee demonstrated the capability to implement the various elements of the dry cask storage program to successfully load and store spent fuel at the ISFSI (Section 1).

Evaluation of General License Requirements

• An extensive review of the licensee's dry cask storage program was completed against the requirements in 10 CFR 72.212 for a general license. The licensee had documented the required evaluations and developed an extensive set of procedures to control work activities associated with the ISFSI. Evaluations had been completed to demonstrate that the design features for the HI-STORM cask system were enveloped by the site specific characteristics of the Columbia Generating Station site (Section 2).

• The licensee conducted a heavy loads movement activity in the train bay using a weighted canister, transfer cask and storage cask. Analysis had determined that modifications to the train bay floor were needed to provide additional support during an earthquake. The licensee performed this activity using the risk assessment techniques allowed for in the new maintenance rule in 10 CFR 50.65 without performing a safety evaluation in accordance with 10 CFR 50.59. This has been determined to be a violation of NRC regulations and is being dispositioned as a Non-Cited Violation (Section 2).

Spent Fuel Pool

• The fuel bridge safety limit controls prevented the grapple from moving too close to the wall of the spent fuel pool and hoses on the grapple prevented the grapple from completely lowering spent fuel assemblies into several locations along the canister wall. The licensee completed modifications to the fuel bridge software and grapple to provide for access to all locations in the canister (Section 4).

• The Part 50 FSAR had included a description of a cask for removing spent fuel from the spent fuel pool. This cask was not approved for storage of spent fuel at an ISFSI and was smaller and lighter than the Holtec design. An amendment to the plant license was required to incorporate the Holtec cask design into the Part 50 FSAR (Section 4).
Procedures and Technical Specification Compliance

• Procedures consisted of a checklist format that provided for good documentation of work activities completed. Procedures included precautions and important reminders of critical parameters. Commitments from the certificate of compliance, technical specifications and the FSAR had been incorporated into procedures. Implementation of

the procedures during the pre-operational tests confirmed the adequacy of the procedures for various work tasks observed during the demonstrations (Section 5).

• The licensee had incorporated written guidance into procedures for abnormal events such as unexpected high dose rates, cask drops, tornado or severe weather conditions, high contamination levels encountered in the work areas, stuck fuel assembly during removal from the fuel racks or during insertion into the cask, and dropped fuel bundles (Section 5).

Safety Reviews

• The licensee had implemented a program to perform safety screenings and evaluations in accordance with the requirements in 10 CFR 50.59 and §72.48. Selected screenings and evaluations performed by the licensee were reviewed and found to be adequately dispositioned (Section 6).

Heavy Loads

• The procedures governing the heavy load lift operations appropriately contained the requirements and guidance from the FSAR and national standards for maintenance and testing to ensure the ability of the equipment to support the anticipated loads required during the dry cask storage program activities (Section 7).

• The heavy loads procedures appropriately addressed cask lift limits. The expected component weights were bounded by the weight values established in the FSAR (Section 7).

• The licensee's planned use of an ancillary device placed under the casks on the 441' elevation and 606' elevation to mitigate the effects of a seismic event were reviewed by the NRC and found acceptable (Section 7).

Training Program

• The licensee had developed a training and certification program for operator personnel performing work at the ISFSI on equipment and controls that were identified as important to safety. The program incorporated the requirements in 10 CFR 72 Subpart I and Section 12.2.1 of the FSAR and included formal classroom training, on-the-job training and specific task demonstrations (Section 13).

• Training was completed for all operator personnel assigned ISFSI duties. Interviews with selected personnel verified that training had been adequately implemented (Section 13).

Quality Assurance Program

• The licensee conducted quality assurance oversight of ISFSI activities using their NRC approved 10 CFR Part 50 quality assurance program. A review of documents, procedures and audits performed by the quality assurance organization determined that the licensee had appropriately applied their Part 50 quality assurance program to the activities associated with the ISFSI (Section 14).

Loading of the First Cask

• The licensee successfully completed the loading of their first cask and placement of the cask on the ISFSI pad on September 20, 2002. Dry cask storage activities were conducted safely and in compliance with procedures. Radiological controls were effectively implemented. The overall dose to complete the project was well below original estimates (Section 17).

From this inspection of the pre-operational testing phase of CGS's ISFSI activities, the NRC determined that a violation of NRC requirements had occurred related to the placement of the cask in the train bay during portions of the pre-operational testing without completing a safety evaluation. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy.

(17) Millstone Units 2 and 3 – NRC Inspection Reports 50-336/02-02 and 50-423/02-02, Dated April 24, 2002. This inspection report documents the NRC's closure of an issue described in LER 50-336/2001-007: Movement of Heavy Loads, as follows. On October 22, 2001, Unit 2 personnel identified an historical issue concerning the movement of heavy loads over a pathway that included a safety related pipe gallery enclosed in a trench. This trench is located below the cask wash down point and the railroad access bay floor. Because the cask crane was not "single failure proof," the licensee performed an analysis and determined that, although the probability was low, a dropped load of sufficient mass could have caused the cask pit floor to fail, resulting in a postulated loss of safety function.

The inspectors verified that the processes controlling such heavy loads were amended to account for this potential. The inspectors coordinated with the Region I, Senior Risk Analyst (SRA) to determine the significance and potential for this postulated event, using the NRC Significance Determination Process.

The inspectors' on-site review and the evaluation of the Region I SRA identified no findings of significance.

(18) Oyster Creek Generating Station – NRC Integrated Inspection Report 50-219/02-02, Dated April 23, 2002. This inspection report documents the NRC's inspection of the pre-operational testing of the ISFSI **(IP 60854)**. The inspectors evaluated whether the licensee was adequately prepared to use the independent spent fuel storage installation (ISFSI). Plans, engineering analyses, work packages, work practices and procedures were reviewed to ensure they met and were consistent with the terms and conditions of the Certificate of Compliance (CoC) for the ISFSI project. The inspectors observed samples of the pre-operational testing of ISFSI operations. This included direct observation of critical activities documented in the CoC Attachment 1, Section 1.1.6, "Pre-Operational Testing and Training Exercises." These activities included, but were not limited to, loading a mock-up fuel assembly into the dry shielded canister (DSC), sealing a mock-up DSC, transfer cask handling, DSC insertion into the horizontal storage module (HSM), and DSC recovery operations.

Selected operational procedures relative to dry cask storage system (DCSS) loading, unloading, and transferring activities were reviewed. The procedures were reviewed to determine if they provided clear instructions to users, established limitations and action levels consistent with CoC requirements and directed workers on what to do if unsafe conditions arose. The acceptance criteria established in procedures were reviewed against requirements and commitments specified in the Safety Analysis Report (SAR), Safety Evaluation Report

(SER), Certificate of Compliance (CoC) (Certificate No. 1004), Standardized Nuclear Horizontal Modular Storage (NUHOMS) System and 10 Code of Federal Regulations (CFR) Part 72. These selected operational procedures were verified to have been prepared, reviewed, and initially approved in accordance with the licensee's administrative programs.

The inspectors reviewed information relative to the licensee's methods for verifying and documenting the parameters and characteristics of spent fuel placed in the dry shielded canister. The review was against criteria contained in the CoC No. 1004, Amendment No. 4, the CoC Technical Specifications and NRC Interim Staff Guidance - 1 (Damaged Fuel). The licensee's 10 CFR 50.59 and 72.48 processes for changes, tests and experiments were reviewed to confirm that a documented and acceptable program was in place for performing design changes or evaluating nonconforming conditions that could affect ISFSI activities. This review included a sample of 10 CFR 72.48 screening/evaluations performed by both the licensee and the CoC holder, Transnuclear West, Incorporated. The inspectors reviewed selected portions of the licensee's 10 CFR 72.212 evaluation, which documented the reviews conducted in accordance with Title 10 Code of Federal Regulations, Part 72, Subpart K, Paragraph 72.212 for the utilization of the NUHOMS 61 BT Dry Spent Fuel Storage System. An evaluation of the ISFSI basemat and approach slab was also reviewed.

The inspectors selectively reviewed radiation protection planning and preparation, radiation work permits, pre-job health physics briefing packages, dose calculations, and the specific radiological hazards identified and the controls to be implemented for the dry cask storage system loading, unloading, and transferring activities. The review was against criteria contained in the CoC No. 1004, Amendment No. 4, the CoC Technical Specifications, 10 CFR 20 and written evaluations required by 10 CFR 72.212. The welding and cutting procedures were reviewed and welding operations performed on the dry shielded canister were observed. The inspectors reviewed the Emergency Plan to ensure that Emergency Action Levels had been developed to address the ISFSI activities. With regard to training, the inspectors selectively reviewed the ISFSI training program and materials. The review verified the CoC requirement that training should include an overview, fuel loading, transfer cask handling, canister transfer procedures and off-normal procedures.

During the Pre-operational testing, lifting of heavy loads was performed. The inspectors reviewed selected documentation for the crane being single-failure proof, including safety evaluations and calculations of maximum loading conditions, applicable controlling procedures, and equipment certifications. This review was performed to ensure the adequacy of rigging, control of heavy loads and crane operations.

The effectiveness of security controls during the pre-operational testing was reviewed. The NRC approved Security Plan and implementing procedures were reviewed to identify the security requirements for the ISFSI and ensure they were being effectively implemented. Testing of the intrusion detection and assessment systems was observed and self audits of ISFSI security implementation were reviewed. Security Management and officials were also interviewed to assure their understanding of the ISFSI security requirements. Additionally, corrective action program issues associated with the ISFSI project were reviewed to ensure that identified deficiencies were properly prioritized for resolution before receipt of fuel at the Independent Spent Fuel Storage Installation.

The inspectors identified no finding from this inspection.

(19) North Anna Power Station – NRC Integrated Inspection Report Nos. 50-338/01-03, 50-

339/01-03 and Independent Spent Fuel Storage Installation Inspection Report No. 72-016/01-01, Dated October 29, 2001. This inspection report documents the NRC's inspection of the transportation of spent fuel cask **(IP 60855, 10 CFR 72 Inspection)** including review of cask crane operating, cask transporting, and transporter operating procedures. The inspectors observed that the licensee transported the loaded cask from the crane bay area to the cask storage area. The inspectors observed the heavy weight truck driving behind the transporter in case of the brake failure on slopes. The inspectors reviewed crane operator training certificates and qualification records.

The inspectors identified no finding from this inspection.

(20) Turkey Point Nuclear Plant – NRC Inspection Report Nos. 50-250/01-09, 50-251/01-09, Dated July 23, 2001. The purpose of this inspection was an examination of activities that support your application for a renewed license for the Turkey Point facilities **(IP 71002)**. The inspection consisted of a selected examination of procedures and representative records, and interviews with personnel regarding the process of scoping and screening plant equipment to select equipment subject to an aging management review. For a sample of plant systems, inspectors performed visual examination of accessible portions of the systems to observe any effects of equipment aging.

The inspection concluded that the scoping and screening portion of your license renewal activities were conducted as described in your License Renewal Application and that documentation supporting your application is in an auditable and retrievable form. With the exception of the items identified in this report, your scoping and screening process was successful in identifying those systems, structures, and commodity groups required to be considered for aging management.

Polar Cranes

The reactor polar cranes and associated rails are seismically qualified Class I structures in the unloaded configuration. The crane provides a means for lifting and handling heavy loads inside the containment structures. The primary components of a polar crane consist of the crane buckets attached to the containment building, the runway rail supported by the crane buckets, the end trucks that ride on the runway rail, the bridge girder that span between the end trucks, the walkway and railing mounted outside one of the girders, the electrical enclosures mounted on the walkway, the cab suspended beneath bridge girder, the trolley rails on top of the bridge girders, and the trolley that rides on the trolley rails.

Engineering document PTN-ENG-LRSP-99-0063 scopes the reactor polar crane as a non safety-related structure whose failure could prevent safety-related systems, structures, or components from performing their intended functions as specified in 10CFR54.4(a)(2). The applicant determined that the reactor polar crane is within the scope of license renewal. The inspectors agreed with the determination.

Section 5.3.3 of PTN-ENG-LRSC-99-0037 states that the evaluation boundary of the reactor polar crane includes everything associated with the polar crane from the crane rail support buckets attached to the containment building to the main and auxiliary hooks, including rails, end trucks, girders, hoists, trolley, and associated sub-components. All the structural components or component types associated with the polar crane are listed in Tables 5.3.1 (Unit 3) and 5.3.2 (Unit 4).

Using the guidance in NEI 95-10, the following components were eliminated from the screening process because they are active components:

- Polar crane end trucks
- Polar crane trolley active sub-components
- Wire rope
- Hook/hook block assembly

The inspectors agreed with the screening results.

Spent Fuel Storage and Handling

Spent fuel is stored in the Unit 3 and Unit 4 spent fuel storage pits. The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies and control rods after removal from the reactor. The pit is lined on the interior surface with stainless steel liner plate. Stainless steel storage racks sitting on the pit floor are provided to hold spent fuel assemblies. The racks 18 are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations.

The spent fuel handling at Turkey Point includes all the equipment and tools necessary to remove spent fuel from the reactor vessel, transport spent fuel to the spent fuel pit, place spent fuel in the appropriate storage rack cell, and remove spent fuel to the spent fuel storage pit for alterative storage. The major equipment required for spent fuel handling includes the reactor cavity seal ring, the manipulator crane, the fuel transfer system, the fuel transfer tube, the spent fuel bridge, the fuel handling tools, and the spent fuel cask crane.

Spent fuel storage includes all the structural components necessary to store spent fuel in the spent fuel storage pit, excluding the concrete structure. The major structural components required for spent fuel storage are the spent fuel pit liner, the keyway gate, and the spent fuel storage racks.

Engineering document PTN-ENG-LRSP-99-0063 lists the spent fuel storage and handling facilities as safety-related structures in Table 2.1-2 and non safety-related structure that could affect safety-related structures in Table 2.2-3, and determines that the spent fuel storage and handling facility is within the scope of license renewal. The inspectors concurred with this decision.

Section 5.4.3 of PTN-ENG-LRSC-99-0037 states the evaluation boundary for the spent fuel storage and handling equipment includes the spent fuel pit area, the transfer canal, the refueling canal, and part of the reactor cavity, plus the cask crane. Using the guidance from NEI 95-10, the applicant determined that the following items are not in scope and thus do not require an aging management review:

- Handling tools - not long lived
- Reactor cavity seal ring inflatable rubber seal - not long lived
- Drive mechanism - active
- Conveyor assemblies and active sub-components - active
- Hoist and active sub-components - active
- Spent fuel pit sliding concrete door and motor - active
- Manipulator crane active features - active

The inspectors agreed with the applicants assessment.

(21) McGuire nuclear Station – NRC Integrated Inspection Report 50-369/00-06 and 50-370/00-06 And Independent Spent Fuel Storage Installation Inspection Report 72-38/00-01, Dated January 12, 2001. This inspection report documents the NRC inspectors review of the ISFSI construction and related plant information **(IP 60851, IP 60854, IP 60855, and IP 60856)**. The reports states inspectors applied **IP 60853** in examining the cask transportation route and facilities to be used for receiving, transferring, and transporting empty or loaded casks from the commercial truck entering the plant boundary to the concrete storage pad. Areas included the turbine building transferring point (which uses a 200 ton Unit 1 turbine building crane for transferring casks), refueling floor, cask decontamination pit, cask loading pit, and the cask storage pads. The inspectors examined the modifications, reinforcement, or markings performed along the route to reinforce the structures or restrict the transporter from running over unsafe or weak areas and compared observations to design drawings. The inspectors also examined the concrete storage pads and related construction.

The inspectors examined the modifications completed around Unit 2 decon pit and platform for the storage of the lifting beam and long pole and compared observations to design drawings. The elements inspected included the member size and lengths, welding sizes and symbols, anchor bolt diameters, and base plate sizes.

The inspectors reviewed an approved procedure and a calculation of the transporting route. The inspectors reviewed maintenance and inspection records for the last two years for the Unit 1 turbine building 200 ton crane and the Unit 2 fuel building 125 ton crane. The inspectors also reviewed records for the transporter load test performed at the manufacturer's facility and transporter route test performed at the site by carrying a 125 ton text weight and driving it on the field route. The inspectors selected and reviewed several Problem Investigation Processes (PIPs).

The inspectors reviewed the licensee's evaluation on the conditions set in Certificate of Compliance for the Transnuclear, Inc., TN-32 Dry Storage Cask (No. 1021). The review included an evaluation to determine compliance with the requirements for operation and design limits set in the general condition for the Technical Specification and special conditions. The inspectors reviewed seven out of ten special conditions which included operation procedures, acceptance test and maintenance program, quality assurance, heavy load requirements, approved contents, design features, and changes to the Certificate of Compliance.

The inspectors verified that the total cask helium leak rate to be less than 1.0 E-5 cc/sec and the helium minimum purity of 99.99% as stated in the procedure MP/0/A/7650/187, Revision 000 were in compliance with Technical Specifications 3.1.3, 3.1.4, and 4.1.4. The licensee evaluated the heavy load requirements for moving the cask from the spent fuel pool area to the concrete storage area by using NUREG-0612, NRC Bulletin 96-02, and ANSI N14.6. The heavy load equipment included the overhead crane, below-the hook spread beam lifting devices, and the crawler type transporter.

The inspectors noted that the minimum cooling time for the spent fuel assemblies after removed from the core and stored in the spent fuel pool shall be more than 7, 8, 9, and 10 years depending on the initial enrichment and burn-up rate, instead of minimum 7 years as stated in the licensee's evaluation.

The licensee calculated the maximum lifting height for the cask moving on the outside of spent

fuel pool area to be 14 inches based on the drop analysis and therefore, limited a lifting height in the operation procedure to be 12 inches maximum from the ground to ensure a margin of safety.

The inspectors identified no finding of significance from this inspection.