UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

October 30, 2019

NRC INFORMATION NOTICE 2019-09: SPENT FUEL CASK MOVEMENT ISSUES

ADDRESSEES

All holders of an operating license or construction permit for a nuclear power reactor issued under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," including those that have permanently ceased operations and have spent fuel in storage in spent fuel pools (SFPs).

All holders of and applicants for a power reactor combined license, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." All applicants for a standard design certification, including such applicants after initial issuance of a design certification rule.

All holders of and applicants for an independent spent fuel storage installation (ISFSI) license under 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of recent issues related to spent fuel cask movement issues. The NRC expects recipients to review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Spent Fuel Cask Load Drop Analysis/Single-Failure-Proof Handling System

San Onofre Nuclear Generating Station

On August 3, 2018, licensee personnel failed to notice that a loaded spent fuel canister was misaligned during a lowering evolution into the vault. The licensee and its contractor continued to lower the vertical cask transporter lift beam until the contractor's staff believed that the canister had been fully lowered to the bottom of the vault. A radiation protection technician identified radiation readings that were not consistent with a fully lowered canister. The licensee then identified that the loaded spent fuel canister was resting on a shield ring near the top of the vault, preventing it from being lowered, and that the rigging and lifting slings were slack and no longer bearing the load of the canister.

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With the slings slack, the lifting equipment was no longer capable of performing its important to safety function of holding and controlling the loaded canister. The canister could have experienced an approximately 17-18 foot drop into the storage vault if the canister had slipped off the shield ring. This condition placed the cannister in an unanalyzed condition because the postulated load drop of a cannister is not a condition analyzed in the dry fuel storage system's Final Safety Analysis Report. The licensee implemented corrective actions that include fuel loading procedural revisions, training of fuel loading personnel and evaluation of any deviations based on cannister contact with vault components for canister integrity.

Additional information appears in "NRC Special Inspection Report 050-00206/2018-005, 050-00361/2018-005, 050-00362/2018-005, 072-00041/2018-001 and Notice of Violation" dated November 28, 2018 (Agencywide Documents and Management System (ADAMS) Accession No. ML18332A357).

Kewaunee Power Station

During an inspection, NRC inspectors reviewed the design qualification of the Secure Lift Yoke/Chain Hoist Assembly used to lift the spent fuel cask. The Updated Safety Analysis Report (USAR) describes the auxiliary building crane as single-failure-proof in accordance with NRC guidance and the cask drop analysis is not part of the licensing basis. The inspectors identified, however, that the Secure Lift Yoke/Chain Hoist Assembly only was qualified as a non-single-failure-proof lifting device to handle a cask containing spent fuel. The non-single-failure-proof lifting device was inconsistent with the licensing basis and created the possibility of dropping a cask, an accident of a different type than described in the USAR, which would require a license amendment pursuant to 10 CFR 50.59. Licensee corrective actions include a license amendment request to use a non-single-failure-proof Secure Lift Yoke/Chain Hoist Assembly as part of cask handling operations within the auxiliary building.

Additional information appears in "NRC Inspection Report No. 050-00305/2015-004(DNMS); 072-00064/2015-002(DNMS) – Kewaunee Power Station," dated August 19, 2016 (ADAMS Accession No. ML16235A301).

Pilgrim Nuclear Power Station

During an inspection at the Pilgrim Nuclear Power Station, the inspectors reviewed the 10 CFR 50.59 regulatory evaluation that removed an energy-absorbing pad from the SFP. This pad was credited for mitigating a postulated spent fuel cask load drop accident. The pad was part of the Technical Specification (TS) requirements since the crane used to lift spent fuel casks was non-single-failure-proof. The licensee installed a single-failure-proof crane, which removed the need for the energy-absorbing pad. Also, the licensee had installed a cask-leveling pad designed to provide protection for the SFP floor liner during cask handling with a single-failure-proof crane, prior to beginning dry storage cask-handling activities. However, the site did not perform an adequate 10 CFR 50.59 regulatory evaluation, which would have concluded that a license amendment was required prior to taking actions that altered the plant from the stated TS condition. Licensee corrective actions included a license amendment request submittal to remove the energy-absorbing pad language from the TS requirement and an extent of condition review on previous engineering changes.

Additional information appears in “Pilgrim Nuclear Power Station NRC Integrated Inspection Report 050-00293/2014-005 and Independent Spent Fuel Storage Installation (ISFSI) Report 072-01044/2014-003,” dated February 4, 2015 (ADAMS Accession No. ML15037A163).

Failure to Follow Boundary Conditions stipulated in ASME NOG-1 2004

Fort Calhoun Station

During an inspection at Fort Calhoun Station, NRC inspectors reviewed a design calculation for the auxiliary building crane, which is classified as seismic Category I. The licensee’s Updated Safety Analysis Report (USAR) specifies the auxiliary building crane meets the requirements of American Society of Mechanical Engineers (ASME) NOG-1-2004, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” as a single-failure-proof system. ASME NOG-1-2004, Section 4153, stipulates the boundary condition requirements for the crane seismic analysis, which delineates full seismic loading at the crane rail/wheel interface. The licensee’s calculation, however, showed that sliding would occur at the crane rail/wheel interface, thus limiting the applied seismic loads to only frictional forces. The inspectors found that the non-linear sliding effects were incorporated in the seismic analysis in a manner inconsistent with the linear elastic analysis methodology. Licensee corrective actions include revising calculations and installing field modifications.

Additional information appears in “Fort Calhoun – NRC Component Design Basis Inspection Report 050-00285/2015-007,” dated April 16, 2015 (ADAMS Accession No. ML15106A891).

Loading on Crane Rail Clip not considered

Clinton Power Station

During an inspection at Clinton Power Station, NRC inspectors reviewed a design calculation for the fuel handling building crane and crane support structure (crane rail clip, rail clip bolts, etc.), which are seismic Category I. The licensee’s USAR specifies the acceptance criteria for Seismic Category I structural steel are based on linear elastic methods and no permanent deformation is allowed. The licensee calculation, however, used the plastic section modulus for the rail clip. The licensee’s USAR specified that Seismic Category I structural steel is designed to the American Institute of Steel Construction specifications. Also, the licensee calculation used friction, bolt preload, and clamping force which resulted in the loading on the rail clip being incorrectly determined and resulted in overestimation of the structural capacity of the rail clip. Licensee corrective actions include calculation revisions and installation of field modifications.

Additional information appears in “NRC Inspection Report Nos. 050-00461/2016-010(DNMS); 072-01046/2016-001(DNMS) – Clinton Power Station,” dated March 3, 2016 (ADAMS Accession No. ML16064A200).

Fort Calhoun Station

During an inspection at the Fort Calhoun Station, NRC inspectors reviewed a design calculation for the auxiliary building crane rail clip. The licensee's USAR specifies that acceptance criteria for safety-related structural steel are based on linear elastic methods and no permanent deformation is allowed. The licensee, however, incorrectly designed the crane runway rail clips to inelastic acceptance limits. ASME NOG-1-2004, Section 4153, stipulates that crane seismic analysis be linear elastic. Instead, the licensee used an allowable bending stress in the calculation consistent with permanent deformation of the rail clip. This assumption resulted in overestimation of the structural capacity of the rail clip. Licensee corrective actions include revising calculations and initiating modifications.

Additional information appears in "Fort Calhoun – NRC Component Design Bases Inspection Report 050-00285/2015-007," dated April 16, 2015 (ADAMS Accession No. ML15106A891).

Inadequate Design of Spent Fuel Cask Laydown Areas

Palisades Nuclear Plant

During an inspection at the Palisades Nuclear Plant, NRC inspectors reviewed design calculations for the stack-up configuration on the auxiliary building trackway slab and identified several issues. First, the inspectors identified that a procedure did not require installation of physical torsional restraints as was assumed in the computer model representing the stack-up configuration. Second, the inspectors identified the interfacing coefficient of friction used in the calculation was based on steel surfaces with an oxide layer consistent with Regulatory Issue Summary (RIS) 2015-13, "Seismic Stability Analysis Methodologies for Spent Fuel Dry Cask Loading Stack-Up Configuration" dated November 12, 2015 (ADAMS Accession No. ML15132A122) guidance. However, the inspectors identified that the installed steel floor plate surface was painted, which could non-conservatively change the interfacing coefficient of friction compared to the evaluated unpainted steel surface. Third, the inspectors identified that, in the field, there was a gap between the components in the stack-up configuration and the analysis did not consider a gap. Lastly, the inspectors identified that the computer analysis results for the truncated low-profile cask transport (used to tow the spent fuel cask) structure with the derived torsional restraint was equivalent to computer analysis results where the entire low-profile cask transport structure was modeled. Therefore, the inspectors determined that the computer results for the analyzed stack-up model with a truncated low-profile cask transport structure were non-conservative. Licensee corrective actions included revising the stack-up seismic analysis to address the identified issues; and translated the analyzed stack-up design configuration into stack-up installation procedures prior to performing stack-up operations with spent nuclear fuel in the multi-purpose canister.

Additional information appears in "Palisades Nuclear Plant - NRC Integrated Inspection Report 050-00255/2016-004; 050-00255/2016-501; 072-00007/2015-001; and 072-00007/2016-001," dated February 14, 2017 (ADAMS Accession No. ML17045A709).

BACKGROUND

Related NRC Generic Communications

IN 2014-12, “Crane and Heavy Lift Issues Identified during NRC Inspection,” dated November 14, 2014 (ADAMS Accession No. ML14149A145). This IN informed addressees of issues identified by NRC inspectors during crane and heavy lift inspections conducted in accordance with guidance from Operating Experience Smart Sample, fiscal year 2007-03, Rev. 2, “Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20,” dated September 12, 2008 (ADAMS Accession No. ML13316C040).

RIS 2005-25, “Clarification of NRC Guidelines for Control of Heavy Loads,” dated October 31, 2005 (ADAMS Accession No. ML052340485). This RIS alerted addressees and clarified guidance related to the control of heavy loads as a result of recommendations developed through Generic Issue 186, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants.”

Supplement 1 to RIS 2005-25, “Clarification of NRC Guidelines for Control of Heavy Loads,” dated May 29, 2007 (ADAMS Accession No. ML071210434). This supplement alerted addressees to the availability of guidance on handling systems, single-failure-proof cranes, and calculational methods for heavy load analyses, as well as communicated regulatory expectations associated with 10 CFR 50.59, “Changes, Tests, and Experiments,” and 10 CFR 50.71(e), as these requirements relate to the safe handling of heavy loads and load drop analyses.

RIS 2005-25 discusses General Design Criterion (GDC) 2, “Design Bases for Protection against Natural Phenomena,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR, Part 50. The RIS specifies, in part, that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes. GDC 4, “Environmental and Dynamic Effects Design Bases,” specifies, in part, that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of missiles that may result from equipment failures.

DISCUSSION

The events above provide examples of issues related to heavy load spent fuel movements. These issues highlight non-compliances with NUREGs, codes, and standards that are part of the plant-specific design and licensing basis.

Although there is no specific requirement to do so, licensees can prevent issues such as those described in this IN by verifying that calculations for load-handling systems and structures designated to support spent fuel casks are consistent with the plant-specific design and licensing bases; and that procedures, training and oversight of spent fuel movement are adequate.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation project manager.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library.

NRC INFORMATION NOTICE 2019-09: FINDINGS RELATED TO SPENT FUEL CASK MOVEMENTS INSIDE THE 10 CFR PART 50 STRUCTURE DATE OCTOBER 30, 2019

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