UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, DC 20555-0001

November 14, 2014

NRC INFORMATION NOTICE 2014-12: CRANE AND HEAVY LIFT ISSUES IDENTIFIED DURING NRC INSPECTIONS

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.

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 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 issues identified by NRC inspectors during crane and heavy lift inspections conducted in accordance with guidance from Operating Experience Smart Sample: (OpESS), fiscal year (FY) 2007–03, Revision 2, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20." OpESS FY 2007-03, Rev. 2, dated September 12, 2008, is available on the NRC's public Web site in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML13316C040. 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

Inappropriate design of crane:

Perry Nuclear Power Plant

During an inspection at the Perry Nuclear Power Plant, NRC inspectors reviewed a design calculation for the containment polar crane, which is classified as seismic Category I.

ML14149A145

The licensee's updated final safety analysis report (UFSAR) specified the crane was designed such that the applied stresses would be less than or equal to the allowable stresses. The crane is designed to maintain the lifted load and structural stability during a Seismic Category I design-basis event. Instead, the licensee's calculation showed the crane trolley seismic restraints had applied stresses greater than the allowable stress for the design-basis seismic loading condition. The licensee performed a walkdown of the restraints to assess the design margin and discovered the restraints were never installed. Licensee corrective actions included installation of crane trolley seismic restraints.

Additional information appears in "Perry Nuclear Power Plant NRC Integrated Inspection Report 05000440/2008005," dated January 26, 2009, on the NRC's public Web site in ADAMS at Accession No. ML090270129.

Duane Arnold Energy Center

During an inspection at the Duane Arnold Energy Center, NRC inspectors reviewed a design calculation for the reactor building crane, which is classified as seismic Category I. The licensee's UFSAR specified that the crane was designed to withstand seismic loads. The design function of the crane is to maintain the lifted load and structural stability during a Seismic Category I design-basis event. Instead, the licensee's calculation did not establish whether the crane rails, which transfer the loads from the crane to the crane support structure, could withstand the seismic loads. Licensee corrective actions included installation of additional crane rails clip restraints to ensure adequate design margin for the crane rail.

Additional information appears in "Duane Arnold Energy Center Integrated Inspection Report 05000331/2010002," dated May 14, 2010, on the NRC's public Web site in ADAMS at Accession No. ML101340444.

Omission of fracture toughness properties in design of special lifting devices

Davis Besse Nuclear Power Station

During an inspection at the Davis-Besse Nuclear Power Station, NRC inspectors reviewed a design calculation for the reactor pressure vessel (RPV) internal handling adapter to demonstrate conformance with American National Standard Institute (ANSI) N14.6-1978, Section 3.2.1, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials," as referenced in the licensee's updated safety analysis report. The adapter pin had a yield strength greater than 80 percent of its ultimate strength, and the licensee did not address material fracture toughness. Instead of performing a brittle fracture analysis, the licensee took corrective actions to replace the pin with a material that had appropriate fracture toughness characteristics.

Additional information appears in "Davis-Besse Nuclear Power Station NRC Integrated Inspection Report 05000346/2007005," dated February 1, 2008, on the NRC's public Web site in ADAMS at Accession No. ML080360447.

Fermi Power Plant

During an inspection at the Fermi Power Plant, NRC inspectors reviewed the design calculation for the steam dryer and steam separator lifting device. The Fermi UFSAR specified the design for this lifting device was to be in accordance with ANSI N14.6-1978. The safety-related function of this lifting device was to hold the heavy load in position during transport and not drop the load onto safety-related systems or irradiated fuel. The inspectors identified that the socket lifting pin and adapter lift pin, which connect the steam dryer and steam separator to the lifting device were not designed based on their material fracture toughness properties as specified in ANSI N14.6-1978, Section 3.2.1.1. Instead, the calculation showed the socket pin and adapter pin had yield strengths greater than 80 percent of their ultimate strengths and the calculation did not address the design based on material fracture toughness characteristics. Licensee corrective actions included performing non-destructive examination of the socket lifting pin and adapter lift pin prior to each use to ensure an adequate factor of safety for brittle fracture.

Additional information appears in “Fermi Power Plant, Unit 2, Integrated Inspection Report 05000341/2012004,” dated October 31, 2012, on the NRC’s public Web site in ADAMS at Accession No. ML12306A184.

Quad Cities Nuclear Power Station

During an inspection at the Quad Cities Nuclear Power Station, NRC inspectors reviewed a design calculation for the steam dryer and steam separator lifting device to demonstrate conformance with ANSI N14.6-1978, Section 3.2.1.1. The inspectors identified that socket pins, lock pins and hook pins were not designed based on their material fracture toughness properties as specified in ANSI N14.6-1978, Section 3.2.1.1. Instead, the calculation showed the socket pins, lock pins and hook pins had yield strengths greater than 80 percent of their ultimate strengths and the calculation did not address the design based on material fracture toughness characteristics. Licensee corrective actions included performing non-destructive examination of the socket pins, lock pins and hook pins prior to each use to ensure an adequate factor of safety for brittle fracture.

Additional information appears in “Quad Cities Nuclear Power Station, Units 1 and 2, Integrated Inspection Report 05000254/2014002; 05000265/2014002,” dated May 9, 2014, on the NRC’s public Web site in ADAMS at Accession No. ML14132A023.

Inappropriate stress design factors of special lifting devices

Duane Arnold Energy Center

During an inspection at the Duane Arnold Energy Center, NRC inspectors reviewed the design calculation for the steam dryer and steam separator lifting device. The Duane Arnold UFSAR specified the design for this lifting device was to be in accordance with ANSI N14.6 1978. The safety-related function of this lifting device was to hold the heavy load in position during transport and not drop the load onto safety-related systems or irradiated fuel. ANSI N14.6-1978, Section 3.2.1.1 specifies that the stress design factors for the load bearing members of the special lifting device be limited to one-third of the yield strength and one-fifth of the ultimate strength. Instead, the licensee used an incorrect shear allowable value for steel members that was not based on one-third of the yield strength. Licensee corrective actions included performing reanalysis to establish the specified design margins for the steam dryer and steam separator lifting device.

Additional information appears in “Duane Arnold Energy Center Integrated Inspection Report 05000331/2010002.”

Fermi Power Plant

During an inspection at the Fermi Power Plant, NRC inspectors reviewed the design calculation for the steam dryer and steam separator lifting device. The Fermi UFSAR specified the design for this lifting device was to be in accordance with ANSI N14.6-1978. ANSI N14.6-1978, Section 3.2.1.1 specifies that the stress design factors for the load bearing members of the special lifting device be limited to one-third of the yield strength and one-fifth of the ultimate strength. Instead, the licensee incorrectly did not include the weight of the lifting device and its intervening components in the determination of applied stress. In addition, the licensee did not analyze all the structural elements in the load path and connection for the stress design factors based on the yield and the ultimate strengths of the material. Licensee corrective actions included performing reanalysis to establish the specified margins for the steam dryer and steam separator lifting device. In addition, NRC reviewed the design calculation for the RPV head strongback and identified similar examples of where stress design factors were not met. The licensee performed reanalysis and identified that the existing RPV head strongback did not meet ANSI N14.6-1978, Section 3.2.1.1. The licensee replaced the existing RPV head strongback with a strongback that met ANSI N14.6-1978, Section 3.2.1.1.

Additional information appears in “Fermi Power Plant, Unit 2 Integrated Inspection Report 05000341/2012004.”

Inadequate licensee inspections of special lifting devices

Perry Nuclear Power Plant

During an inspection at the Perry Nuclear Power Plant, NRC inspectors reviewed the Control of Heavy Loads licensing basis for the RPV head strongback which is a special lifting device. The Perry USAR specified that the periodic testing for the special lifting devices used for safe transport of heavy loads, specifically that the RPV head strongback was in accordance with ANSI N14.6-1978. Section 5.3.1 of ANSI N14.6-1978, specifies that each special lifting device be subjected to either a load test or dimensional testing, visual inspection, and nondestructive testing of major load carrying welds and critical areas. The NRC inspectors identified that the licensee had not performed nondestructive testing of the load carrying welds or a load test. Licensee corrective actions included performing, prior to each use, dimensional testing, visual inspection, and nondestructive testing of major load carrying welds and critical areas of the RPV head strongback.

Additional information appears in “Perry Nuclear Power Plant Integrated Inspection Report 05000440/2008005.”

Fermi Power Plant

During an inspection at the Fermi Power Plant, NRC inspectors reviewed the licensing basis regarding licensee inspections of the RPV head strongback and the steam dryer/steam separator lifting device. The function of the strongback and lifting device is to support the lifted load and safely transport the reactor pressure head, over the reactor pressure vessel and over safety-related equipment to the designated laydown area during a refueling outage. The licensee's UFSAR specified that periodic testing for the lifting devices was to be in accordance with ANSI N14.6-1978, Section 5.3.1, which specifies that each special lifting device be subjected to either a load test or dimensional testing, visual inspection, and nondestructive testing of major load carrying welds and critical areas. The NRC inspectors identified that the licensee did not perform dimensional testing of the RPV head strongback and steam dryer and steam separator lifting device, nondestructive testing of major load carrying welds and critical areas of the steam dryer and steam separator lifting device, or a load test on either. Licensee corrective actions included performing, prior to each use, dimensional testing, visual inspection, and nondestructive testing of major load carrying welds and critical areas of the RPV head strongback and steam dryer/steam separator lifting device.

Additional information appears in "Fermi Power Plant, Unit 2 Integrated Inspection Report 05000341/2012004."

BACKGROUND

Related NRC Generic Communications

NRC Regulatory Issue Summary (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), 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.

DISCUSSION

This IN provides examples of issues identified by NRC inspectors during crane and heavy lift inspections conducted in accordance with OpESS FY 2007-03, Rev. 2. The NRC inspection findings involve instances of not having adequate stress design factors, not performing adequate nondestructive examination of load carrying welds, and not designing the lifting device components for material fracture toughness. The safety implications of these issues is a reduction in load handling reliability of the lifting device and increase in the likelihood of a structural component failure that could result in a load drop that could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The safety implication of a crane and its components not designed to withstand the design loading conditions is that the crane or crane components could lack structural integrity and result in crane structural damage, which, in turn, could adversely impact structures, systems, or components.

Although there is no specific requirement to do so, licensees can prevent non-conformances such as those described in the IN by verifying that calculations for crane and special lifting devices satisfy the codes and standards referenced in applicable licensing and design bases. Licensee can also verify that the procedures used to implement load testing or visual testing, dimensional testing, nondestructive examination of major load carrying welds, and critical areas for the special lifting devices satisfy the codes and standards referenced in applicable licensing and design bases.

RIS 2005-25 provides references regarding the design of cranes and control of heavy loads used in the establishment of plant specific design and licensing bases. 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. It 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,” of Appendix A to 10 CFR Part 50 specifies, in part, that structures, systems, and components important to safety shall be appropriately protected against dynamic effects, including the effects of missiles that may result from equipment failures. The guidelines of NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants” and NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” were developed to implement these criteria in the design of overhead heavy load handling systems.

In addition, RIS 2005-25 discusses that in NUREG-0612, the NRC staff provides regulatory guidelines for the control of heavy loads in areas where a load drop could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In a letter dated December 22, 1980, later identified as Generic Letter (GL) 80-113, as supplemented by GL 81-07, “Control of Heavy Loads,” dated February 3, 1981, the NRC staff requested that all licensees describe how they satisfied the guidelines of NUREG-0612 at their facilities and what additional modifications would be necessary to fully satisfy these guidelines. The NRC staff divided this request into two phases (Phase I and Phase II) for implementation by licensees. Phase I guidelines addressed measures for reducing the likelihood of dropping heavy loads and provided criteria for establishing safe load paths; procedures for load handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and selection and use of slings. Phase II guidelines addressed alternatives to reduce further the probability of a load handling accident or mitigate the consequences of heavy load drops. These alternatives include using a single-failure-proof crane for increased handling system reliability, employing electrical interlocks and mechanical stops for restricting crane travel to safe areas, or performing load drop and consequence analyses for assessing the impact of dropped loads on plant safety and operations. In NUREG-0554, the NRC staff included the criteria for the design of single-failure-proof cranes. In Appendix C to NUREG-0612, NRC staff provided alternative criteria for upgrading the reliability of existing cranes to single-failure-proof standards.

NUREG-0612, Section 5.1.1(4), "Special lifting devices," states that special lifting devices should satisfy the guidelines of ANSI N14.6-1978. ANSI N14.6 can be used to establish design, testing, and inspection adequacy for a lifting device used to carry heavy loads over or near irradiated fuel or safe shutdown equipment. Each reactor licensee will have a plant specific licensing basis for the special lifting devices that was established based on NRC staff review using NUREG-0612, Section 5.1.1(4). Licensees that intend to have Independent Spent Fuel Storage Installations at their reactor site will determine the crane and special lifting device requirements for the spent nuclear fuel dry storage cask handling system as defined in the spent nuclear fuel dry storage cask licensing documents, including the technical specifications.

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 NRC 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.

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