UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
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NRC REGULATORY ISSUE SUMMARY 2005-25, SUPPLEMENT 1 CLARIFICATION OF NRC GUIDELINES FOR CONTROL OF HEAVY LOADS

ADDRESSEES

All holders of operating licenses for nuclear power reactors.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing Supplement 1 to Regulatory Issue Summary (RIS) 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads," issued October 31, 2005, to 1) announce the availability of guidance on handling systems, single failure proof cranes and calculational methods for heavy load analyses, and 2) communicate regulatory expectations associated with Title 10 of the *Code of Federal Regulations* (10 CFR) Sections 50.59 and 50.71(e), as these requirements relate to the safe handling of heavy loads and load drop analyses. This RIS requires no action or written response on the part of the addressees.

BACKGROUND

The NRC issued RIS 2005-25 principally to reemphasize the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," issued July 1980 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070250180) and inform industry of relevant operating experience. In issuing RIS 2005-25 on October 31, 2005, the NRC began to address recommendations identified through the investigation of Generic Issue (GI) 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Since the issuance of RIS 2005-25, NRC inspectors have continued to identify heavy load movements performed outside bounds confirmed by analyses to be safe. In issuing this supplement, the NRC addresses remaining recommendations associated with GI 186 and communicates regulatory expectations related to safe load handling.

In NUREG-0612, the NRC staff provided regulatory guidelines for the control of heavy loads to assure the safe handling 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 (ADAMS Accession No. ML071080219), as modified by GL 81-07 (ADAMS Accession No. ML031080524), "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 facility 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. The NRC staff requested Phase I responses within 6 months that addressed the following elements:

- the extent of heavy load handling in areas where spent fuel is stored, in the vicinity of the reactor vessel, and in other areas where a load drop could result in damage to equipment required for safe shutdown or decay heat removal,
- the establishment of safe load paths for these heavy load handling evolutions,
- the development of procedures for load handling operations,
- the training of crane operators,
- the design, testing, inspection, and maintenance of cranes and special lifting devices, and
- the selection and use of slings.

The NRC staff requested Phase II responses within 9 months that addressed alternatives to either reduce further the probability of a load handling accident or demonstrate the consequences of heavy load drops were within acceptable limits. These alternatives include using a single-failure-proof crane to increase handling system reliability, employing electrical interlocks or mechanical stops to restrict crane travel to safe areas, or performing load drop and consequence analyses for assessing the impact of dropped loads on plant safety. The NRC staff requested that load drop and consequence analyses conform to the guidelines of Appendix A, “Analyses of Postulated Load Drops,” to NUREG-0612. These guidelines specified assumptions to use in the analyses, including the following:

- The load is dropped in the orientation that causes the most severe consequences.
- The load may be dropped at any location in the crane travel area where mechanical stops or electrical interlocks do not restrict movement.
- All potential accident cases during the refueling operation should be considered, including the fall of the reactor vessel head from its maximum height.
- All potential accident cases should be analyzed for the actual medium present (e.g., water in the spent fuel pool or above the reactor vessel when the refueling cavity is flooded, and air when the refueling cavity is drained).
- Energy-absorbing devices integral to fuel casks may be credited if attached during handling operations.

Application of these assumptions to load drop analyses essentially results in analyses of limiting load drops in that the heaviest load is assumed to drop in the most damaging orientation from the greatest potential height in the location resulting in the greatest potential damage.

The Phase I and Phase II responses established the bases for heavy load handling programs at nuclear power plants. During the review of these responses, the NRC staff requested additional information about issues such as safe load paths, special lifting devices, crane design, and special compensatory measures during certain load handling evolutions. Licensees generally incorporated the information contained in the initial and supplemental responses into the heavy load handling program described in each facility's safety analysis report. However, NRC inspections have revealed that additional clarification regarding the proper licensing treatment of the load drop and consequence analyses would be beneficial.

In GL 85-11 (ADAMS Accession No. ML031150689), "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612," dated June 28, 1985, the NRC staff concluded that a detailed review of the Phase II responses received from licensees was not necessary. Specifically, the letter stated the following:

> All licensees have completed the requirement to perform a review and submit a Phase I and a Phase II report. Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action is not required to reduce the risks associated with the handling of heavy loads (See enclosed NUREG-0612 Phase II). Therefore, a detailed Phase II review of heavy loads is not necessary and Phase II is considered completed. However, while not a requirement, we encourage the implementation of any actions you identified in Phase II regarding the handling of heavy loads that you consider appropriate.

The NRC staff based its conclusion on the improvements resulting from the review of the Phase I responses and the findings identified through a pilot review of several Phase II responses as well as a more limited review of the remaining Phase II responses. The pilot review was based on Phase II responses from 20 operating reactors at 12 sites and 6 operating license applicants at 5 sites. Of those 26 reactors, all 10 boiling-water reactors had single-failure-proof cranes, 10 pressurized-water reactors (PWRs) had load drop analyses demonstrating satisfactory outcomes, and 6 PWRs had a combination of administrative controls and limited load drop analyses demonstrating satisfactory outcomes. This latter approach did not meet the guidelines of Appendix A to NUREG-0612 because licensees changed important assumptions associated with the load drop analyses. These changes were associated with administrative controls related to the handling of heavy loads. Enclosure 1 to GL 85-11 described the limited load drop analyses and administrative controls as an *amplification of* the guidelines of the Phase I effort. Specifically, the analyses define restrictions that are incorporated into safe load paths and load handling procedures.

As documented in Enclosure 1 to GL 85-11, the NRC staff found no heavy loads handling concerns of sufficient significance existed to justify further generic action. Plants that had installed single-failure-proof cranes or completed load drop analyses conforming with the guidelines of Appendix A to NUREG-0612 would remain in conformance with Phase II guidelines, absent a physical change to the facility. The NRC staff considered the administrative controls related to limited load drop analyses as an amplification of Phase I guidelines regarding procedures for the safe handling of heavy loads. The NRC staff found that the administrative controls implemented in load handling procedures provided an acceptable level of safety and further enhancements to conform with Phase II guidelines would not provide a substantial improvement in safety. Accordingly, the NRC staff stated in GL 85-11 that a detailed Phase II review of heavy loads is not necessary and Phase II is considered completed.

SUMMARY OF ISSUE

Availability of Guidance Documents

Rigging Used with Single-Failure-Proof Handling Systems

As documented in NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," issued July 2003, below-the-hook events have increased and contributed to a number of load drops. Recent operating experience includes several events in which synthetic round slings have failed because of inadequate corner protection, resulting in drops of loads exceeding 30 tons in weight. The NRC staff has not identified operating experience indicating that metallic slings have failed in use at a similar rate based on information contained in NUREG-1774. The NRC staff found the single operational errors that resulted in synthetic round sling failures to be incompatible with the intent of single-failure-proof handling systems. Operating experience suggests that metallic slings are more resistant to similar load handling errors.

In Revision 1 to Section 9.1.5, "Overhead Heavy Load Handling System" (ADAMS Accession No. ML062260190), of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter, referred to as the Standard Review Plan (SRP)), the NRC staff revised the guidelines for the use of lifting devices with single-failure-proof cranes. The change specified the use of slings constructed from metallic material where the single-failure-proof features of the handling system are credited in achieving a very low probability of a load drop. Specifically, the SRP called for the selection of lifting devices for use with single-failure-proof handling systems that satisfy either of the following two criteria:

(1) A special lifting device that satisfies American National Standards Institute (ANSI) N14.6-1993, "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," should be used for recurrent load movements in critical areas (i.e., reactor heads, reactor vessel internals, spent fuel casks). The lifting device should have either dual, independent load paths or a single load path with twice the design safety factor specified by ANSI N14.6 for the load.

(2) Slings should satisfy the criteria of American Society of Mechanical Engineers (ASME) B30.9-2003, "Slings," and be constructed of metallic material (chain or wire rope). The slings should be either (a) configured to provide dual or redundant load paths or (b) selected to support a load twice the weight of the handled load.

This change in guidance does not require any action for currently licensed reactors. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required.

ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes"

In Revision 1 to Section 9.1.5 of the SRP (NUREG-0800), the NRC staff also enhanced the guidelines for the design of single-failure-proof cranes. The NRC staff has concluded that the application of the criteria for Type 1 cranes from ASME NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes," to the design of new overhead heavy load handling systems is an acceptable method for satisfying the guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

These guidelines provide licensees and applicants with guidance that the NRC considers acceptable for the use in designing new single-failure-proof cranes. The NRC staff has participated in the ASME Cranes for Nuclear Facilities Committee as it has developed a comparison of ASME NOG-1 design criteria to the criteria of NUREG-0554. The NRC staff understands that the committee will provide the comparison as an appendix to a future revision of ASME NOG-1.

Calculational Methodologies for Heavy Load Drop Analyses

As documented in NUREG-1774, the NRC staff identified that calculational methodologies, assumptions, and predicted consequences varied greatly from licensee to licensee for very similar load drop accident scenarios. The load drop analyses considered by the NRC staff principally involved drops onto concrete floor slabs. Accurate load drop analysis is essential, since licensees use load drop calculations to determine the transport height restrictions referenced in its heavy load lift procedures. Load drop analyses also help to determine locations where other measures besides load height restrictions are necessary (e.g., impact limiting devices, interlocks to prevent crane motion over certain areas, or employment of single-failure proof handling systems). The NRC staff evaluated the need to establish standardized calculation methodologies for heavy load drops, and concluded that existing NRC guidance and an industry consensus standard provide adequate guidance for evaluation of heavy load drops.

In Revision 2 to Section 3.8.4, "Other Seismic Category I Structures," of the SRP (NUREG-0800), the NRC staff presented a list of codes and standards applicable to design and analysis procedures used for Seismic Category I structures. The NRC staff listed American Concrete Institute (ACI) standard, ACI-349-1997, "Code Requirements for Nuclear Safety Related Concrete Structures," with supplemental guidance from Regulatory Guide (RG) 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," for concrete structures and ANSI/American Institute of Steel Construction (AISC) standard N690-1994, including Supplement 2 (2004), "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," for steel structures.

The standard ACI-349-1997 is an industry consensus document that provides criteria for the design of safety related concrete structures. In November 2001, the NRC issued Revision 2 to RG 1.142, which provides guidance on the implementation of ACI 349-1997 in the design of safety-related concrete structures. Regulatory Positions 10 and 11 of RG 1.142 endorse Appendix C, "Special Provisions for Impulsive and Impactive Effects," of ACI 349-1997, with certain clarifying exceptions that reflect existing review practices of the NRC staff (e.g., an exception that limits ductility ratio for some load cases and restricts the use of a dynamic increase factor for a dynamic load factor in cases where the structure is found to be responding in a static or semi-static manner to a dynamic load). The provisions of Appendix C to ACI 349-1997 apply to those structural elements directly affected by the impactive and impulsive loads where failure of the structural element must be precluded. Impactive loads to be considered under Appendix C to ACI 349 include fuel cask drops and other internal and external missiles. Accordingly, the NRC staff concluded that a standardized calculational methodology for analyses of load drops onto concrete floor slabs is available. Similarly, ANSI/AISC N690-1994 provides calculational methodology for analyses of load drops onto steel structures.

Regulatory Expectations Associated with 10 CFR 50.59 and 50.71(e)

10 CFR 50.71(e) requires that licensees revise the updated final safety analysis report (FSAR) to include the effects of the following:

- all changes made in the facility or procedures as described in the FSAR,

- all safety analyses and evaluations performed by the licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment, and

- all analyses of new safety issues performed by or on behalf of the licensee at Commission request.

The effects of these changes includes appropriate revisions of descriptions in the FSAR such that the FSAR (as updated) is complete and accurate.

Many licensees addressed the safety issues associated with heavy load handling through installation of a single-failure-proof crane. Licensees that installed single-failure-proof cranes prior to 2000 typically made that change in the facility in association with a license amendment request; therefore, 10 CFR 50.71(e) requires that licensees incorporate the change in the FSAR. In addition, licensees have installed single-failure-proof cranes to handle new loads in support of conclusions that changes in heavy load handling did not require a license ammendment. Licensees must also incorporate these changes in the FSAR.

Alternatively, many licensees evaluated the consequences of heavy load drops in areas where heavy load drops could damage spent fuel or equipment required for safe reactor shutdown or decay heat removal. Licensees concluded that the consequences of postulated load drops satisfied acceptance criteria described in the NRC staff request for information in GL 80-113. In support of these evaluations, licensees performed site-specific load drop analyses or referenced generic load drop analyses for similar configurations. These analyses included parameter values and assumptions, such as lift height, load weight and medium present under the load. These parameter values and assumptions are material to the evaluation conclusion and define restrictions on the motion of heavy loads necessary to demonstrate that the acceptance criteria were satisfied. The consequence evaluations of the (then) new safety issues associated with heavy load drops were performed at Commission request in response to GL 80-113. Therefore, 10 CFR 50.71(e) requires that licensees incorporate in the FSAR a description of the consequence evaluation and elements of the underlying analyses necessary to make the description complete and accurate. The motion restrictions, such as lift height, load weight and medium present under the load to cushion postulated drops, are appropriately within the scope of procedures governing load handling.

10 CFR 50.59 contains requirements for the process by which licensees may make changes to their facilities and procedures as described in the FSAR without prior NRC approval, under certain conditions. Procedures that contain information described in the FSAR such as how structures, systems and components are operated and controlled are subject to the requirements of 10 CFR 50.59. 10 CFR 50.59(c)(2) describes 8 conditions that would require licensees to obtain a license amendment prior to implementation of a proposed change. One specific condition described is 10 CFR 50.59(c)(2)(v), that would require a license amendment for a change that would create a possibility for an accident of a different type than any previously evaluated in the FSAR (as updated). As a result, changes to the procedures described in the FSAR governing load handling, such as changes to motion restrictions, maximum heights and material present under the loads to cushion drops, are subject to the requirements of 10 CFR 50.59 to determine if the change meets one of the conditions described in 10 CFR 50.59(c)(2) including creating the possibility for an accident of a different type. RG 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," November 2000, and Nuclear Energy Institute (NEI) 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," November 2000, provide guidance for developing effective and consistent 10 CFR 50.59 implementation processes.

BACKFIT DISCUSSION

This RIS requires no action or written response. Any action by addressees to implement changes is strictly voluntary and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because the RIS is informational and pertains to a staff position that does not depart from current regulatory requirements and practices.

CONGRESSIONAL REVIEW ACT

This RIS is not a rule as designated by the Congressional Review Act (5 U.S.C. §§ 801-808) and, therefore, is not subject to the Act.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain any information collections and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

CONTACT

Please direct any questions about this matter to the technical contact listed below.

/RA/

Michael Case, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Technical Contacts: Steven R. Jones, NRR
301-415-2712
E-mail: srj@nrc.gov

Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

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