U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.244, Revision 0

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CONTROL OF HEAVY LOADS AT NUCLEAR FACILITIES

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) to meet regulatory requirements for the control of heavy loads at nuclear facilities. Specifically, these requirements call for licensees to provide appropriate protection against equipment failure that could result in a heavy load drop. This RG endorses the following, with clarifications:

- American Society of Mechanical Engineers (ASME) Standard (Std.) NML-1–2019, "Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities," June 28, 2019 (Ref. 1)

- ASME Std. NOG-1–2020, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," December 4, 2020 (Ref. 2)

- ASME Std. BTH-1–2017, "Design of Below-the-Hook Lifting Devices," March 15, 2017, Chapters 1–3 (Ref. 3)

The NRC staff expects endorsement of these consensus standards to provide safety and efficiency benefits.

Applicability

This RG applies to applicants and licensees subject to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 4); to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," for a power reactor combined license (Ref. 5); and to 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" (Ref. 6), for an independent spent fuel storage installation (ISFSI). This RG also applies, in part, to

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Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML21006A346. The regulatory analysis may be found in ADAMS under Accession No. ML21006A337. The associated draft guide DG-1381 may be found in ADAMS under Accession No. ML21006A335, and the staff responses to the public comments on DG-1381 may be found under ADAMS Accession No. ML21244A455.

applicants and holders of standard design certifications, standard design approvals, or manufacturing licenses for power reactors under 10 CFR Part 52.

Applicable Regulations

The regulations in 10 CFR Part 50 establish criteria for the licensing of production and utilization facilities, including nuclear power plants. The specific sections applicable to this RG include the following:

- 10 CFR 50.34, "Contents of application; technical information," requires, in part, that power reactor license applicants include in the safety analysis report (1) the design of the facility (including the principal design criteria, the design bases, and the relationship of the design bases to the principal design criteria), (2) a description and analysis of auxiliary and fuel handling systems insofar as they are pertinent to showing that safety functions will be accomplished, and (3) a description of plans for the conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components (SSCs). The description should be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the principal design criteria for nuclear power plants. The general design criteria (GDC) in 10 CFR Part 50, Appendix A, applicable to this RG include the following:

    - GDC 1, "Quality standards and records," requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

    - GDC 2, "Design bases for protection against natural phenomena," requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, with appropriate consideration in the design basis of (1) the most severe natural phenomena historically reported, (2) combinations of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions threatened by the natural phenomena.

    - GDC 4, "Environmental and dynamic effects design bases," requires appropriate protection for SSCs important to safety against dynamic effects, including the effects of missiles (e.g., falling heavy loads) that may result from equipment failures.

The regulations in 10 CFR Part 52 establish criteria for the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. To that end, the specific 10 CFR Part 52 regulations applicable to this RG include the following:

- 10 CFR 52.47, "Contents of applications; technical information"; 10 CFR 52.137, "Contents of applications; technical information"; and 10 CFR 52.157, "Contents of applications; technical information in final safety analysis report," require, in part, that applicants for nuclear power facility standard design certifications, standard design approvals, and manufacturing licenses, respectively, include in the safety analysis report (1) the design of the facility (including the principal design criteria, the design bases, and the relationship of the design bases to the principal design criteria) and (2) a description and analysis of auxiliary and fuel handling systems insofar as they are pertinent to showing that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

- 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," requires, in part, that power reactor combined license applicants include in the safety analysis report (1) the design of the facility (including the principal design criteria, the design bases, and the relationship of the design bases to the principal design criteria), (2) a description and analysis of auxiliary and fuel handling systems insofar as they are pertinent to showing that safety functions will be accomplished, and (3) a description of plans for the conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

The regulations in 10 CFR Part 72 establish criteria for the issuance of specific or general licenses to receive, transfer, and possess power reactor spent fuel, power reactor-related greater than Class C waste, and other radioactive materials associated with spent fuel storage in an ISFSI. To that end, the specific 10 CFR Part 72 regulations applicable to this RG include the following:

- 10 CFR72.24, "Contents of application: Technical information," requires, in part, that applicants for ISFSIs include in the application (1) the design of the facility (including design criteria, the design bases, materials of construction, and applicable codes and standards), (2) an evaluation of the design of SSCs important to safety with regard to their performance in preventing or mitigating accidents, and (3) a description of the quality assurance program that satisfies the requirements of 10 CFR Part 72, Subpart G, "Quality Assurance," applied to the design, fabrication, construction, testing, and operation of SSCs important to safety.

- 10 CFR 72.122, "Overall requirements," requires, in part, that SSCs important to safety be designed to withstand postulated accidents, such as handling system component failures, and with appropriate consideration of the effects of natural phenomena.

Related Guidance

- RG 1.13, "Spent Fuel Storage Facility Design Basis" (Ref. 7), provides guidance specifying that cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving over the spent fuel storage pool.

- NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," issued July 1980 (Ref. 8), provides criteria for the protection of critical SSCs from the effects of heavy load handling system failures and specifies good standard industrial practices for the handling of heavy loads.

- NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," issued May 1979 (Ref. 9), provides technical guidance for the design, fabrication, installation, and testing of overhead cranes with the ability to withstand credible component failures, natural phenomena, and operator errors while maintaining control of the suspended load.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50, 52,and 72 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0011, 3150-0151, and 3150-0132. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch ((T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011, 3150-0151, and 3150-0132), Office of Management and Budget, Washington, DC, 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

B. DISCUSSION

Reason for Issuance

The NRC staff is issuing the RG for the purpose of endorsing the following three ASME standards as stated:

(1) ASME Std. NML-1–2019 provides programmatic guidance and is endorsed with clarifications that specify information needed to demonstrate that safety functions would be appropriately protected against handling system equipment failures. The NRC endorsement of ASME Std. NML-1–2019 in this RG updates the guidance in NUREG-0612.

(2) ASME Std. NOG-1–2020 is the crane design standard and is endorsed with clarifications that specify information needed related to the application of quality assurance measures and the interface of the crane with other structures. ASME Std. NOG-1–2020 is used to design overhead cranes with multiple girders and top running trollies using wire rope hoists. The NRC endorsement of ASME Std. NOG-1–2020 in this RG updates the guidance in NUREG-0554.

(3) ASME Std. BTH-1–2017 provides criteria for the design of special lifting devices and load lifting attachments. The staff endorses this standard in part, limiting the endorsed scope to that for mechanical special lifting devices conforming to Design Categories B and C as defined in the standard and excluding sections that address specialized equipment (e.g., electrical components, vacuum lifting devices, and electromagnetic lifting devices). This specialized equipment would not be acceptable for the principal safety function of retaining the suspended load, but specialized equipment may be mounted on a special lifting device for other purposes.

The use of consensus standards where available is consistent with Commission policy and provides updated information reflecting operating experience and risk-informed considerations. The existing technical reports, NUREG-0612, issued August 1980, and NUREG-0554, issued May 1979, do not reflect current technologies and risk-informed perspectives for heavy load handling activities.

Background

The staff recognizes that safe control of heavy load handling activities in nuclear facilities may be accomplished in several ways. The preferred method, as suggested in RG 1.13, is to design the layout of the facility so that overhead lifting equipment cannot operate over or near SSCs essential to the accomplishment of fundamental safety functions.[1] However, light-water reactor design and operation involve certain load handling activities, such as transfer of irradiated fuel from storage pools to dry storage and removal of the reactor vessel head and internal structures in support of refueling, that, if the load were dropped, could challenge the performance of safety functions. Other reactor types may also require load handling activities for continued operation that could similarly challenge safety functions. In addition, existing facilities were constructed without full consideration of load handling activities that could challenge safety functions in other ways. For these handling evolutions, the staff has accepted other methods of providing reasonable assurance that key safety functions would be accomplished in the event of failures affecting load handling equipment.

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1 For the purpose of this RG, the fundamental safety functions are (1) control of nuclear reactivity, (2) adequate removal of heat from the reactor and from stored irradiated fuel, (3) appropriate confinement of radioactive material, and (4) maintenance of adequate shielding against radiation.

The staff described these methods in NUREG-0612, which provides for the use of one of the following methods in areas where a handling system failure could challenge the execution of key safety functions: use of a highly reliable handling system, verification by analysis that safety functions would be accomplished in the event of a load drop, or the use of electrical interlocks or mechanical stops to prevent the motion of loads over SSCs necessary to perform key safety functions. The guidelines in NUREG-0612 establish the following criteria to demonstrate by analyzing load drop consequences that safety functions had been accomplished:

- Releases of radioactive material result in doses well within regulatory limits.

- Damage does not result in a fuel configuration with an effective neutron multiplication factor greater than 0.95.

- Resulting leakage from the reactor vessel and spent fuel pool is within makeup capabilities.

- At least one train of equipment performing essential safety functions would be undamaged.

The guidelines of NUREG-0612 established specific criteria for a highly reliable (i.e., single-failure-proof) handling system. This system consisted of (1) an overhead crane designed to the criteria of NUREG-0554, (2) either lifting devices that are not specially designed (i.e., slings) or a special lifting device designed to American National Standards Institute (ANSI) Std. N14.6-1978, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More for Nuclear Materials," February 15, 1978 (Ref. 10), configured in either balanced, redundant paths with normal design factors or in a single path with twice the normal design factor, and (3) interfacing lift points (i.e., lifting attachments such as lifting lugs or trunnions) configured for either lifting devices with balanced, redundant paths having normal design factors or for lifting devices with a single path having twice the normal design factor.

In addition, the guidelines of NUREG-0612 provide practices addressing the conduct of normal load handling operations, including maintenance and testing. These practices include (1) defined load paths, (2) load handling procedures, (3) trained crane operators, (4) special lifting devices designed and maintained to appropriate standards, (5) appropriately selected standard lifting devices, (6) inspection, testing, and maintenance of the overhead crane to appropriate standards, and (7) overhead cranes designed to appropriate minimum standards.

In 2019, ASME issued Std. NML-1–2019 to provide guidelines for conducting lifting and handling operations at nuclear facilities using overhead handling systems. Compared to NUREG-0612 guidelines, the standard covers a broader scope in terms of the types of overhead handling systems and the safety significance of the load handling activities. The NUREG-0612 guidelines applied only to inherently stable overhead cranes with top running trolleys and wire rope hoists used in areas around stored nuclear fuel or over SSCs essential to achieve and maintain safe shutdown. The guidelines of Std. NML-1–2019 apply to a variety of additional overhead handling systems, including gantries with hydraulic jacking towers commonly used for handling dry spent fuel storage cask system components, underhung hoists suspended from monorails or jib cranes commonly used for heavy-component movement inside nuclear facility buildings, and mobile cranes with extendable booms for movement of heavy components above and around smaller structures (e.g., service water pump replacement through the roof of a water intake structure).

The guidelines of Std. NML-1–2019 use a risk-informed approach by qualitatively considering the probability and consequences of a load handling event during a planned lift to accommodate variations in safety significance and the associated controls applied to manage the risk of the lift. The probability of a load handling event is established by evaluating the handling system design, rigging configuration, and load characteristics. The potential consequences of a load handling event are classified considering the properties of the lifted load and the SSCs in the vicinity of the load path. The classifications are "standard lift," "special lift," and "critical lift," with "nuclear safety critical lift" a subset of critical lift. Nuclear safety critical lifts encompass lifts that could challenge the continued performance of fundamental safety functions and are comparable to the scope of lifts the staff considered in developing NUREG-0612. Maintenance activities in which a load handling event could reduce the redundancy of equipment available to perform a fundamental safety function but could not directly prevent the accomplishment of the function may be classified as special lifts.

The guidelines of ASME Std. NML-1–2019 address the plans for conducting normal handling system operations, including maintenance, surveillance, and periodic testing of handling system SSCs. These plans for normal operations include lift procedures; crane operator and rigger qualifications; crane inspection, testing, and maintenance; and selection and use of lifting devices not specially designed. These guidelines have been expanded to address the variety of load handling systems that may be used for safety-significant load handling evolutions. These guidelines provide an acceptable means of defining plans for conducting normal heavy load handling activities consistent with applicable regulations.

ASME Std. NML-1–2019 also establishes criteria for protecting safety functions during nuclear safety-critical lifts. These criteria include control of load motion by design and interlocks, enhanced safety handling systems, and engineering controls that include analysis of a postulated load drop. The enhanced safety handling system includes a crane designed as single-failure proof, such as an overhead crane designed to meet the Type I criteria of ASME Std. NOG-1–2020, and lifting devices, which may include special lifting devices designed to ASME Std. BTH-1–2017 for recurrent lifts. The acceptance criteria for load drop analyses remain the same as those in the NUREG-0612 guidelines in all material aspects. The ASME Std. NML-1–2019 guidelines for establishing enhanced safety handling systems using an overhead crane designed to meet the Type I criteria of ASME Std. NOG-1–2020 in its entirety constitutes a complete, acceptable method of evaluation to demonstrate that safety functions would be accomplished with appropriate consideration of the effects of natural phenomena or credible equipment failures. The ASME Std. NML-1–2019 guidelines for controlled ranges of motion, as clarified in this guide, also constitute a complete, acceptable method of evaluation to demonstrate that safety functions would be accomplished with appropriate consideration of the effects of natural phenomena or credible equipment failures. Thus, adoption of these guidelines (i.e., ASME Std. NML-1 guidelines for enhanced safety handling systems and controlled ranges of motion for nuclear safety critical lifts) as clarified in this guide would likely not result in a conclusion that it was a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses, as defined in 10 CFR 50.59 or 10 CFR 72.48. However, the guidelines in ASME Std. NML-1–2019 for enhanced safety handling systems using cranes of other designs, as addressed in this guide, or for establishing engineering controls derived from the analysis of postulated load drops provide an acceptable approach for developing a method of evaluation demonstrating that safety functions would be accomplished. Licensees are responsible for evaluating changes to operating procedures and the facility, as described in the safety analysis report, pursuant to 10 CFR 50.59 or 10 CFR 72.48, but the staff expects application of ASME Std. NML-1–2019 guidelines to load handling activities would not involve adverse changes to the likelihood or consequences of load handling activities because the standard provides for good quality procedures, training, equipment, maintenance, and in-service inspection and testing.

The NRC staff defined criteria for cranes used as part of a highly reliable handling system in NUREG-0554. This NUREG provides technical guidance for the design, fabrication, installation, and testing of overhead cranes with the ability to withstand credible component failures, natural phenomena, and operator errors while maintaining control of the suspended load. The specifications for a Type I crane in ASME Std. NOG-1–2020 provide a more comprehensive standard for the design, fabrication, and preoperational testing of these cranes that better incorporates knowledge from crane manufacturers, crane users, and the NRC staff, among other stakeholders, as part of the consensus standard development process.

The NRC staff previously endorsed criteria contained in ANSI Std. N14.6–1978 for design and testing to demonstrate the continued compliance of special lifting devices. However, the issuing organization has withdrawn the standard, and the staff has found the design and testing criteria to be overly conservative for devices intended for routine use at a single nuclear facility because the standard was intended for equipment routinely transported between facilities and frequently used. The guidelines of ASME Std. NML-1–2019 reference ASME Std. BTH-1–2017 for the design and fabrication of special lifting devices and specify a special test program to demonstrate continued compliance of the special lifting device with the intended design margin of safety. Accordingly, the NRC staff elected to endorse the design and fabrication specified in ASME Std. BTH-1–2017, in combination with the specific testing to monitor the quality of the special lifting devices provided in ASME-Std. NML-1–2019, to replace comparable information contained in ANSI Std. N14.6–1978.

Many operating reactor licensees have committed to using the guidelines of NUREG-0612 for heavy load handling activities. Section 2-6, "Nuclear Safety Critical Lifts," paragraph (b), of ASME Std. NML-1–2019 states that facilities with a control of the heavy load handling program described in the facility safety analysis report may continue to handle nuclear safety critical lifts in a manner consistent with the control of the heavy loads program. Nonmandatory Appendix A to ASME Std. NML-1–2019 includes a conformance matrix comparing the guidelines of NUREG-0612 with the guidelines from ASME Std. NML-1–2019. Similarly, nonmandatory Appendix C to ASME Std. NOG-1–2020 includes a conformance matrix comparing the guidelines of NUREG-0554 with the overhead crane design specifications in ASME Std. NOG-1–2020. These appendices serve as an aid to licensees that want to modify their licensing basis to use ASME Std. NML-1–2019 or ASME Std. NOG-1–2020, or both, to control heavy loads in accordance with the requirements of 10 CFR 50.59, "Changes, tests and experiments." However, consistent with NRC backfit and issue finality policies, the continued use of existing heavy load handling programs or the development of new heavy load handling programs based on the provisions of NUREG-0612 and NUREG-0554 remains acceptable.

Similarly, the licenses for ISFSIs (specific or general) have invoked the guidelines of NUREG-0612 for handling loaded canisters. These license requirements typically specify the use of handling systems, including cranes, that satisfy the criteria of NUREG-0554 unless the loaded canister and any surrounding overpack have been designed to withstand postulated drops that encompass proposed handling operations. As stated above, the continued use of existing heavy load handling programs or the development of new heavy load handling programs based on the provisions of NUREG-0612 and NUREG-0554 remains acceptable.

The staff has issued modifications and clarifications to regulatory guidance related to the control of heavy load handling activities in Regulatory Issue Summary (RIS) 05-025, "Clarification of NRC Guidelines for Control of Heavy Loads," dated October 31, 2005 (Ref. 11); RIS-05-025, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads," dated May 29, 2007 (Ref. 12); and RIS-08-28, "Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts," dated December 1, 2008 (Ref. 13). In RIS-05-025, the staff discusses the relationship between heavy load handling guidelines and nuclear power plant operating experience. The NRC issued Supplement 1 to RIS-05-025 to notify stakeholders of modifications to heavy load handling regulatory guidance for the NRC staff that were derived from operating experience. Separately, the NRC issued RIS-08-028 to endorse initiative guidelines developed by the Nuclear Energy Institute that were intended for voluntary implementation by licensees to ensure that heavy load lifts continue to be conducted safely and that the plant licensing bases accurately reflect facility heavy load handling practices. These guidance documents continue to be relevant to those facilities that have incorporated the regulatory information into their facility licensing basis. However, ASME Std. NML-1–2019 included changes to heavy load handling guidance derived from operating experience and an allowance for the incorporation of existing heavy load handling program elements adopted through implementation of the Nuclear Energy Institute initiative. A specific example of operating experience transferred to guidance in ASME Std. NML-1–2019 relates to sling usage where synthetic slings used in a basket configuration for lifts of loads with angular edges resulted in load drops when sling protection did not remain in the necessary position. Therefore, licensees that choose to adopt the guidelines of ASME Std. NML-1–2019 have implicitly adopted the guidance associated with the NUREGs and generic communications discussed above and further consideration of their content during license applications and other change processes is not necessary.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops safety requirements and safety guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA safety requirements and safety guides pursuant to the Commission's International Policy Statement (Ref. 14), and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 15).

The NRC considered the following IAEA Safety Requirement in the update of this RG:

- IAEA Specific Safety Requirements (SSR)-2/1, "Safety of Nuclear Power Plants: Design," Revision 1, issued February 2016 (Ref. 16), addresses design considerations for overhead lifting equipment in nuclear power plants in Section 6, "Design of Specific Plant Systems," as "Requirement 76: Overhead Lifting Equipment," and in associated paragraph 6.55. This RG incorporates similar design guidelines and is consistent with the fundamental safety principles in IAEA SSR-2/1.

Documents Discussed in Staff Regulatory Guidance

This RG endorses the use of one or more codes or standards developed by external organizations, and other third party guidance documents. These codes, standards and third party guidance documents may contain references to other codes, standards or third party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a RG, then the secondary reference is neither a legally-binding requirement nor a "generic" NRC approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

The NRC staff is issuing this RG for the purpose of endorsing the following three ASME standards, with the clarifications stated below:

(1) ASME Std. NML-1–2019, "Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities,"

(2) ASME Std. NOG-1–2020, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," and

(3) ASME Std. BTH-1–2017, "Design of Below-the-Hook Lifting Devices," in part, only Chapters 1 through 3.

Regulatory Position C.1

1. This RG endorses ASME Std. NML-1–2019 as acceptable guidance for establishing a nuclear facility heavy load handling program, subject to the following clarifications:

a. In regard to Section 2-6.1(c)(1) of ASME Std. NML-1–2019, if the handling-system-controlled range of motion is credited to ensure that essential safety functions would be maintained, then the associated crane should meet the criteria for an ASME Std. NOG-1–2020, Type II, crane (or another type of crane designed to maintain its position under seismic loading, consistent with the facility licensing basis, to preclude the potential for the crane itself to fall) and the evaluation of handling system controls for range of motion should include the following considerations:

(1) Boundaries of the range of motion should include the following specific factors:

- a margin for tipping equal to the height of the load when the characteristics of planned handling system lifts include a load with a center of gravity higher than the width of its base in its planned handling orientation, and

- exclusion of components from within the range of motion when secondary effects from that component's failure (e.g., flooding from pressurized piping that could credibly cause failure of equipment outside the boundary) would prevent performance of an essential safety function.

(2) Administrative measures should prevent unintended disabling of the range of motion controls when they are not a permanent feature of the design (e.g., key-operated bypass of electrical motion interlocks).

b. In regard to Section 2-6.1(c)(2) of ASME Std. NML-1–2019, if enhanced handling system reliability is credited to ensure essential safety functions would be maintained, then the handling system design should meet the crane and lifting device standards by conforming to the following guidelines:

(1) The crane should meet one of the endorsed standards as described in Regulatory Position C.2, unless one of the conditions listed in (2) applies.

(2) If the proposed nuclear safety critical lift is—

- outside of nuclear power plant structures (e.g., operations related to an independent spent fuel storage facility),

- involves an infrequent major component replacement (i.e., large components whose handling is not described in the facility safety analysis report), or

- inside nuclear power plant structures with inadequate space or strength to accommodate a crane conforming to Regulatory Position C.2,

then the alternative lifting system designs should either be previously accepted by the NRC staff for the intended application, consistent with the critical lift guidelines of Section 4-1.1, "Overhead Crane," of ASME Std. NML-1–2019, or the approach to the design should address the following criteria:

(a) Meet applicable national consensus standard(s) identified within Section 1-2, "Scope," of ASME Std. NML-1–2019 for the type(s) of lifting system(s) used.

(b) Apply quality assurance in design, fabrication, installation, and initial testing commensurate with the component's importance to stopping or holding the load.

(c) Apply conservative design criteria to the structural elements essential to support the load (e.g., apply structural load combinations and design criteria consistent with those specified in the standards cited in Regulatory Position C.2).

(d) Apply conservative design criteria to the mechanical components essential to stopping or holding the load (e.g., apply mechanical design criteria consistent with those specified in the standards cited in Regulatory Position C.2).

(e) Include redundancy in the design of mechanical components essential to stopping or holding the load that are subject to fatigue or wear (i.e., active components that rotate or change configuration to accomplish a function necessary to stop or hold the load).

(f) Use fail-safe electrical systems and components when failure of the electrical system or component could affect the ability to stop or hold the load.

(g) Apply appropriate quality assurance measures to electrical components intended to detect equipment failures and that actuate following those equipment failures to stop or hold the load.

(3) Special lifting devices and load lifting attachments should meet the endorsed standard as described in Regulatory Position C.3.

(4) Standard lifting devices (e.g., slings) and lifting hardware (e.g., shackles) should meet the selection criteria and use restrictions specified in Sections 5-1.2.1, "Considerations for Critical Lifts," and 5-2, "Other Rigging Equipment," of ASME Std. NML-1–2019, with the exception that the provision of Section 5-1.2.1(b) limiting sling usage configurations (i.e., direct, straight-line connections between hardware or lifting attachments designed for slings or slings used in a basket configuration when the lifted load provides a smooth rounded surface for the sling having a diameter at least 25 times the sling diameter) for synthetic slings applies regardless of sling material.

c. In regard to Section 2-6.1(c)(3) of ASME Std. NML-1–2019, the following clarifications apply to analyses of postulated load drops with the potential to damage multiple irradiated fuel assemblies:

(1) The analysis of radiological consequences should conform to appropriate guidance from RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” (Ref. 17).

(2) The consequences of the postulated drop should not result in deformation or repositioning of the fuel pins that would preclude modeling of the fuel as a regular array for evaluation of nuclear reactivity.

d. In regard to Section 2-6.2, “Movement of Loads over Irradiated Fuel,” of ASME Std. NML-1–2019, the staff endorses the use of an enhanced reliability handling system that satisfies Regulatory Position C.1.b for handling heavy loads directly over irradiated fuel, as the preferred means of ensuring safety functions would be accomplished, or, as a secondary option, the use of engineering controls as described in Section 2-6.1(c)(3) of ASME Std. NML-1–2019 and clarified by Regulatory Position C.1.c.

Regulatory Position C.2

2. This RG endorses ASME Std. NOG-1–2020 criteria for a Type I crane as acceptable guidance for cranes with multiple girders, a top running trolley, and a wire rope hoist used as part of an enhanced reliability handling system, subject to the following clarifications:

a. A quality assurance program equivalent to the applicant’s or licensee’s quality assurance program should be applied to the design, fabrication, installation, and initial testing of the crane. The scope of the quality assurance program for an ASME Std. NOG-1–2020, Type I, crane should encompass those components defined in Section 2000, “Quality Assurance,” of ASME Std. NOG-1–2020 for a Type I crane.

b. The crane runway or other structural support is outside the scope of ASME Std. NOG-1–2020, and, therefore, the adequacy of the runway or structure to support the transmitted load for the load combinations included in the crane design should be confirmed, consistent with the facility design basis for structures important to safety.

Regulatory Position C.3

3. This RG endorses the guidelines of Chapters 1 through 3 of ASME Std. BTH-1-2017 for design and fabrication of special lifting devices and load lifting attachments that form a mechanical load path from the load to the crane hook (or other designed attachment point) and are used as part of an enhanced reliability handling system. The specific design criteria are subject to the following clarifications:

a. Design Category B applies to lifting devices configured to provide two independent load paths between the load and the crane attachment points in a manner such that each load path can independently hold the load in a stable configuration.

b. Design Category C applies to lifting devices configured to provide a single load path between the load and the crane attachment point.

c. Load lifting attachments for both standard lifting devices and special lifting devices should be designed to Design Category B criteria for two independent load paths and Design Category C criteria for single load path configurations.

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, “Backfitting”; 10 CFR 70.76, “Backfitting”; and 10 CFR 72.62, “Backfitting”; and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” (Ref. 18), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

REFERENCES

1. American Society of Mechanical Engineers (ASME), Standard NML-1-2019, "Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities," New York, NY, June 28, 2019.[2]

2. ASME, Standard NOG-1-2020, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," New York, NY, December 4, 2020.

3. ASME, Standard BTH-1-2017, "Design of Below-the-Hook Lifting Devices," New York, NY, March 15, 2017.

4. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."[3]

5. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

6. CFR, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," Part 72, Chapter 1, Title 10, "Energy."

7. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis."

8. NRC, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980. Agencywide Documents Access and Management System (ADAMS) Accession No. ML070250180.

9. NRC, NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979. ADAMS Accession No. ML110450636.

10. American National Standards Institute (ANSI) Std. N14.6-1978, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More for Nuclear Materials," New York, NY, February 15, 1978.

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[2] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, New York 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/

[3] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed on line or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

11. NRC, Regulatory Issue Summary (RIS) 05-025, "Clarification of NRC Guidelines for Control of Heavy Loads," October 31, 2005. ADAMS Accession No. ML052340485.

12. NRC, RIS 05-025, Supplement 1,"Clarification of NRC Guidelines for Control of Heavy Loads," May 29, 2007. ADAMS Accession No. ML071210434.

13. NRC, RIS 08-28, "Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts," December 1, 2008. ADAMS Accession No. ML082460291.

14. NRC, "Nuclear Regulatory Commission International Policy Statement," *Federal Register*, Vol. 79, No. 132, pp. 39415–39418 (79 FR 39415), Washington, DC, July 10, 2014.

15. NRC, Management Directive 6.6, "Regulatory Guides," Washington, DC.

16. International Atomic Energy Agency (IAEA), "Safety of Nuclear Power Plants: Design," Specific Safety Requirements (SSR) 2/1, Rev. 1, Vienna, Austria, February 2016.[4]

17. NRC, RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

18. NRC, Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," September 20, 2019. ADAMS Accession No. ML18093B087.

[4] Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria.