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NRC REGULATORY ISSUE SUMMARY 2005-25: 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 this regulatory issue summary (RIS) to clarify 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," and findings developed through the NRC inspection program.

BACKGROUND INFORMATION

General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," 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 for implementation of these criteria in the design of overhead heavy load handling systems.

The guidelines in NUREG-0612 minimize the occurrence of the principal causes of load handling accidents and provide an adequate level of defense-in-depth for the handling of heavy loads near spent fuel and safe shutdown systems. Defense-in-depth is generally defined as a set of successive measures that reduce the probability of accidents or the consequences of such accidents. In the control of heavy loads, the NRC staff emphasizes measures that prevent load drops or other load handling accidents. If analyses demonstrate acceptable consequences from potential load drop accidents, licensees can use these analyses as an acceptable means of achieving defense-in-depth.

In NUREG-0612, the NRC staff provides 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

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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 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. 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.

The responses to GL 81-07 established the bases for heavy load handling programs at nuclear power plants. During the review of the responses to GL 81-07, 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 the facilities safety analysis reports.

In GL 85-11, “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. 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 of several Phase II responses. The pilot 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 (BWRs) 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. Based on these reviews, the NRC staff concluded that the cost to install single-failure-proof polar cranes in PWR containment buildings was not justified on a generic basis. Nevertheless, the NRC staff encouraged licensees to determine and implement the appropriate actions to provide adequate safety.

In NRC Bulletin 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment,” dated April 11, 1996, the staff addressed concerns on specific instances of heavy load handling and requested that licensees provide information documenting their compliance with these guidelines and their licensing bases.

The heavy load handling concerns were principally related to the increasing frequency of movement of 90 metric ton (100 U.S. ton) or greater spent fuel storage casks during power operation as independent spent fuel storage installations have been licensed at operating reactor sites. To investigate the need for additional regulation or guidance to address the risk with these heavy load movements, the Office of Nuclear Regulatory Research (RES) accepted this concern as Generic Issue (GI) 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants."

The survey of operating experience performed as part of the investigation of GI-186 was documented in NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002." In evaluating operating experience data collected for NUREG-1774, the staff determined that the frequency of load drops was low and unlikely to justify additional regulations or guidance. The survey of operating experience identified some problems that could be addressed by clarification and reemphasis of existing guidance.

NRC is issuing this RIS to address the following recommendations endorsed by the Advisory Committee on Reactor Safeguards (ACRS) in a letter dated September 23, 2003 (ML032681205):

- Reemphasize the need to follow NUREG-0612 guidelines, which address good practices for crane operations and load movements, and continue to assess implementation of heavy load controls in safety-significant applications through the Reactor Oversight Process.

- Evaluate the capability of rigging components and materials to withstand rigging errors (e.g., absence of corner softening material, acute angle lifts, shock from load shifts, and postulated human errors).

- Evaluate the need to establish standardized calculation methodologies for heavy load drops.

- Endorse American Society of Mechanical Engineers (ASME) NOG-1, "Rules for Construction of Overhead and Gantry Cranes" for Type 1 cranes.

This RIS does not issue new guidelines for rigging applications or crane operations. Rather, it identifies operating experience and inspection information related to the movement of heavy loads. The staff uses this information to reemphasize the general guidelines of NUREG-0612 in Section A of Attachment 1. Section A.5 of Attachment 1 describes operating experience information related to slings and other rigging components and discusses specific rigging components and materials that have failed in recent industry applications. The staff has identified operating experience and inspection findings related to load drop analyses in Section B.1 of Attachment 1. Any standardization of load drop analyses methodologies would be accomplished through development of consensus standards with industry. The NRC staff is participating on the ASME Cranes for Nuclear Facilities Committee to support endorsement of ASME NOG-1 for the design of new single-failure-proof cranes in a future supplement to this RIS. However, the staff describes issues related to upgrading of existing cranes to single-failure-proof designs in Section B.2 of Attachment 1. The staff also describes NRC regulations that may be applicable to changes in the heavy load handling program, including how those requirements relate to crane upgrades, in Section C of Attachment 1.

SUMMARY OF ISSUE

Heavy load handling at nuclear power plants may involve risk to stored irradiated fuel and to equipment necessary for a safe shutdown of the reactor. Although the estimated frequency of heavy load drops is low, there is considerable uncertainty when determining the risk of heavy load movement. Drop frequency is highly dependent on human performance, and it is difficult to identify safe shutdown systems that may be affected by potential load drops. Therefore, the staff is clarifying and reemphasizing existing regulatory guidelines that enhance human performance or compensate for human performance errors. Many of these guidelines have been incorporated in site-specific heavy load programs described in the facility's safety analysis report.

Heavy component movements within the reactor building while fuel is in the reactor vessel and spent fuel cask movements have the highest potential risk. Because of plant arrangement, heavy load drops in BWR plants with Mark I or Mark II containments are more risk significant overall than heavy load drops in PWR plants or BWR plants with a Mark III containment. For PWR plants and BWR plants with a Mark III containment, spent fuel cask movement typically occurs in an area separate from the reactor building and systems essential to place the plant in a safe shutdown condition. Heavy load movement within the reactor building at these plants is typically limited to heavy component movement associated with refueling and maintenance when the plant is shutdown. Therefore, the principal safety concerns related to heavy load handling at these plants involve load drops that damage either the spent fuel storage facilities, fuel in the reactor vessel, or the residual heat removal capability (including the reactor coolant system) while the plant is shutdown with fuel in the reactor vessel. For BWR plants with a Mark I or Mark II containment, many heavy loads (e.g., spent fuel casks and drywell shield blocks) are lifted and moved on the upper floor of the reactor building while the reactor is operating at power. If a floor breach were to occur during a load drop, safety-related components located on the lower floors could be adversely affected. A load drop that penetrates the operating floor in certain areas could simultaneously initiate an accident and disable equipment necessary to mitigate the accident.

In Section 5.1 of NUREG-0612, the NRC staff provides recommended guidelines to preserve defense-in-depth for the handling of heavy loads at nuclear power plants. Licensees can achieve defense-in-depth by implementing measures that both reduce the probability of a load drop and the probability that a dropped load could damage reactor fuel or equipment essential for a safe shutdown. To reduce the probability that a load, if dropped, could damage irradiated fuel or safe shutdown equipment, licensees can implement the safe load paths and load handling procedures in Section 5.1.1 of NUREG-0612. Licensees can also reduce the probability of a load drop by improving the reliability of the handling system components through design, operation, maintenance, and inspection of cranes and associated lifting devices to appropriate standards, as described in Section 5.1.1 of NUREG-0612.

In Sections 5.1.2–5.1.5 of NUREG-0612, the NRC staff describes measures that provide defense-in-depth for specific areas within nuclear power plants. These measures include the following:

- mechanical stops and electrical interlocks that prevent heavy load movement over irradiated fuel or safe shutdown equipment;
- verification analysis that the consequences of a potential load drop are within acceptable bounds; or
- use of a single-failure-proof handling system.

In Section 5.1.6 of NUREG-0612, the NRC staff defines the criteria for licensees to implement a single-failure-proof handling system and references NUREG-0554 for crane design criteria and Appendix C to NUREG-0612 for guidelines to upgrade existing cranes.

Attachment 1 to this RIS provides descriptions of insights gained from operating experience and inspection into application of the guidelines of NUREG-0612 to heavy load handling at nuclear power plants. In Attachment 1, the staff also provides clarification of the guidelines where operating experience or inspection results indicated that clarification was necessary.

Heavy load handling activities do pose a safety risk in areas of nuclear power plants where load drops could damage irradiated fuel or equipment necessary for safe shutdown. The NRC staff developed the guidelines in Section 5.1 of NUREG-0612 to reduce the frequency of heavy load drops at nuclear power plants and provide a measure of defense-in-depth. The defense-in-depth measures provide assurance that the probability of a load handling event that damages irradiated fuel or safe shutdown equipment is acceptably small. Licensees incorporated many of these recommendations into site-specific heavy load handling programs. However, operating experience and inspection findings related to heavy load handling indicate that additional clarification and reemphasis of heavy load handling guidelines may reduce the frequency of load handling events.

BACKFIT DISCUSSION

This RIS requires no action or written response and is, therefore, not a backfit under 10 CFR 50.109. Consequently, the staff did not perform a backfit analysis.

FEDERAL REGISTRATION NOTICE

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

SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT of 1996

The NRC has determined that this action is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain 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

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Attachment: Clarification and Reemphasis of Guidelines for Control of Heavy Loads

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

CLARIFICATION AND REEMPHASIS OF GUIDELINES FOR THE CONTROL OF HEAVY LOADS

INTRODUCTION

The NRC staff initiated Generic Issue 186, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,” to investigate the need for additional regulation or guidance to address the risk with the increased frequency of heavy load movements associated with the dry storage of spent fuel and major component replacement activities. NRC documented its investigation in NUREG-1774, “A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002.”

This survey indicated that there was an increase in load drop events involving overhead cranes similar to those used in safety-related areas of the power plant. When compared to the previous period from 1981 to 1992, the period from 1993 to 2002 experienced a 60% increase in the number of load drop events, concurrent with an increase in the number of operating units by 9% (documented in NUREG-1774). The number of below-the-hook crane events (mainly rigging deficiencies or failures) has increased greatly. For the period from 1968 to 2002, there were 47 reported below-the-hook events, many resulting in load drops and damaged equipment. Over the last decade (1993–2002), there were 33 below-the-hook events, of which 17 involved load drops, 10 involved equipment damage, 4 involved administrative issues, and 2 involved load slips. This represents an increase in the number of below-the-hook crane events by 230% compared to the previous decade. Although the data include events associated with cranes other than the overhead cranes typically used in areas containing safety-related equipment, several significant events involved overhead cranes.

In addition, the survey of operating experience found that the calculational methodologies, assumptions, and predicted consequences of load drop events varied greatly from licensee to licensee for very similar accident scenarios. Accurate load drop analyses are essential, because licensees use load drop calculations to determine the lift height restrictions referenced in their heavy load 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).

Although the survey indicated that the overall frequency of heavy load drops remained low, the strong contribution of human performance errors to identified load handling events indicated that the frequency estimate is subject to substantial uncertainties. Therefore, the NRC staff concluded that identification of heavy load handling program problems and clarification and reemphasis of existing guidelines would be appropriate actions to reduce the uncertainty in human performance.

The following sections describe measures commonly incorporated in heavy load handling programs to either reduce the probability of load drop events or evaluate the consequences of such events. This section reemphasizes existing commitments typically contained in heavy load handling programs and clarifies where the guidelines incorporated into heavy load programs have not been correctly implemented as indicated by operating experience and inspection findings. This discussion does not change each operating reactor's specific commitments related to heavy load handling.

DISCUSSION

A. GENERAL GUIDELINES (Section 5.1.1 OF NUREG-0612)

1. Safe Load Paths

Safe load paths should be defined in operating procedures and operator training such that, to the extent practicable, heavy loads are carried over neither irradiated fuel nor equipment necessary to safely shutdown the plant and maintain it in a safe shutdown condition. Compliance with these guidelines provides a measure of defense-in-depth; because, in the event of a heavy load drop, the nuclear safety consequences of the drop are less likely to be significant.

In some cases, the orientation of the load is important in complying with these guidelines. At Palisades, the orientation of the load was improper and caused a portion of a reactor coolant pump motor to be carried over irradiated fuel seated in the reactor vessel. If the licensee had aligned the load with the direction of travel, the motor could have been moved without passing over irradiated fuel. This issue is documented in NRC Inspection Report 05000255/2004012 (ML050320365), dated January 31, 2005.

Although NUREG-0612 suggests that the load path follow structural floor members to the extent practicable, advanced analysis of floor systems indicates that open floor spans may have greater capability to absorb the energy of a load drop without damage than floor sections directly over structural members. Floor spans with this capability have adequate reinforcement to absorb the tensile and shear stresses induced by a load drop in an elastic manner and distribute the energy to multiple structural members. In some cases, a drop of the load directly over a structural support exceeds the energy absorption capability of that member because the floor reacts in a less flexible manner and the energy of the drop is concentrated on that single structural member.

2. Procedures for Load Handling Operations

In NUREG-0612, the NRC staff advised licensees to include the following in their procedures:

- identification of required equipment,
- inspections and acceptance required prior to movement of the load,
- the steps and proper sequence to be followed in handling the load,
- definition of the safe load path, and
- other special precautions.

In NUREG-1774, the failure to properly implement procedural requirements was identified as the principal contributor to load handling events, including load drops. In particular, improper implementation of procedural requirements for the selection and use of slings contributed to several load drops.

Operational experience indicates that "below-the-hook" human performance problems are the principal contributor to load drops. Since the ability to compensate for these errors is limited, human performance in this area is important and the measures that offer protection against these errors, such as increased safety factors used in the selection of slings, should have the expected capability to compensate for likely human performance errors. Several recent heavy load drops have involved sling failure due to inadequate protection of slings used in a basket configuration (see Section 5, "Slings," below). Slings used in basket[1] and choker[2] configurations are vulnerable to failure through cutting because the sling bears against the load at corners. Operator performance in correctly placing protective pads at these bearing locations or selecting alternative sling configurations that mitigate the risk of cutting is important in preventing load drops.

3. Crane Operators

In NUREG-0612, the NRC staff endorsed ANSI-B30.2-1976, "Overhead and Gantry Cranes." The guidelines of NUREG-0612 state that crane operators should be trained, qualified and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976. This standard has been generally incorporated into each plant's heavy load handling program. Although newer editions of this standard have been issued, the NRC staff has not formally endorsed them.

ANSI B30.2-1976 includes a provision for special heavy lifts. The special heavy lift provision applies to special purposes such as new construction or major repairs, and includes a requirement that structural, mechanical, and electrical components of the crane be checked by a crane manufacturer or other qualified person to an accepted crane design standard such as CMAA-70.

Newer editions of ANSI-B30.2 include provisions for an "engineered lift." An engineered lift is an infrequent lift requirement exceeding the design rated load of the crane. The rated load is used in the design of the crane to verify the structures and components satisfy the design criteria of CMAA-70, "Specifications for Electric Overhead Traveling Cranes," or other standard accepted for crane construction. The NRC staff considers use of the engineered-lift provision appropriate for major component replacement at nuclear power plants when the lift does not pose a nuclear safety concern (e.g., all irradiated fuel has been removed from a PWR containment prior to movement of a replacement steam generator within containment).

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[1] a sling configuration where both ends of the sling are attached to the crane hook and the body of the sling directly supports the load.

[2] a sling configuration where the sling wraps around the load and one end of the sling passes through an eye or loop at the other end of the sling before attaching to the crane hook.

4. Special Lifting Devices

Section 5.1.1 of NUREG-0612 states that special lifting devices should satisfy the guidelines of ANSI N14.6, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500 kg) or More for Nuclear Materials." Special lifting devices designed to ANSI N14.6-1976 have proven to be effective in preventing serious load handling events. Heavy load handling programs have typically included these devices for routinely handled heavy loads, such as reactor vessel heads, vessel internals, and reactor coolant pump motors. Errors in the use of these handling devices have involved failure to properly insert pins and other mechanical fasteners at connections between load-bearing members. Although these failures have resulted in tilted or uneven loading of the lifting device, these errors have not resulted in identified load drops.

5. Slings

The guidelines of NUREG-0612 state that lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings." This section also states that design of lifting devices and selection of slings should be based on the combined maximum static and dynamic loads, excluding the loads imposed by the safe shutdown earthquake. This standard has been generally incorporated into each plant's heavy load handling program, and the heavy load handling programs may identify that dynamic loading be considered in the selection of slings. Although newer editions of this standard have been issued, NRC has not formally endorsed the newer editions. The 1971 version of this standard addressed the use of slings constructed from chain, wire rope, synthetic rope, or synthetic webbing.

Subsequent editions of the standard have addressed the use of synthetic round slings[3]. These slings offer improvements in ease of handling for heavy loads. However, these slings are relatively easily damaged, especially when compared to chain or wire rope slings. The majority of below-the-hook events identified in NUREG-1774 involve failure of these synthetic round slings, and these failures have been the result of inadequately protecting the slings from damage in basket sling configurations. Examples of these failures include a 12-meter (40-foot) drop of a 34 metric ton (37.5 U.S. ton) mobile crane at San Onofre Unit 3 that severely damaged the mobile crane and a short drop of a similar mobile crane at Turkey Point. In addition, the drop of a 30 metric ton (32.5 U.S. ton) overhead crane trolley[4], in the process of being lowered for replacement at Browns Ferry, damaged the refueling floor in the defueled Unit 1 reactor building. Details of this event are described in NRC Integrated Inspection Report 05000260/2004005 (ML050310001), dated January 28, 2005. These load drops occurred in

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[3] a rigging device configured as a loop and constructed from continuous synthetic (e.g., polyester, nylon, or aramid) fibers sheathed in a durable, woven cover.

[4] a structural unit that travels on the main bridge rails and houses the hoisting machinery of the crane.

plant areas where damage to irradiated fuel or safe shutdown equipment was not a concern, but they exemplify the potential for a single error to result in a load drop that causes substantial damage to structures or components.

The guidelines contained in NUREG-0612 do not include provisions for use of intermediate hoists between the hook from the overhead crane and the special lifting device or sling. Intermediate hoists increase the potential for a load drop because the hoists are typically designed to less rigorous standards than the overhead crane hoist. Also, the intermediate chain hoists include many individual components whose failure would result in a drop of the load. This potential was illustrated by the 6- to 9-meter (20- to 30-foot) uncontrolled lowering of a 38 metric ton (42 U.S. ton) reactor coolant pump motor at Comanche Peak Nuclear Power Plant. The gear train of the 41 metric ton (45 U.S. ton) rated intermediate hoist failed, which allowed the chain to move freely through the hoist. Only a fortuitous snag of a hoist chain link in the hoist load block prevented probable impact of the motor on the reactor coolant pump base and reactor coolant system piping. There was no nuclear safety concern because all irradiated fuel had been transferred to the spent fuel pool prior to the lift. This event is documented in NRC Inspection Report 05000445/99-16 (ML993620200), dated December 10, 1999.

A related issue was identified at the South Texas Project, Unit 1, and was documented in NRC Inspection Report 05000498/2003002 (ML032170569), dated August 5, 2003. The licensee employed an intermediate hoist for lifting a reactor coolant pump motor from its base because the main hook was too large for the loop area and overhead clearance was inadequate to use a lifting device extension. In Mode 5 (cold shutdown), the licensee lifted the approximately 45 metric ton (50 U.S. ton) motor over the in-service residual heat removal heat exchangers, which are located within containment, as well as over the pump base, which is part of the reactor coolant system. The licensee’s heavy load program specified that “when heavy loads are carried over an RHR train with less than a 10:1 interface lift points safety factor, the RHR train shall be declared inoperable and isolated from the reactor coolant system (RCS) prior to moving the load over the RHR train.” However, the chain hoist used to lift the motor provided only a 5:1 safety factor (commercial grade lifting equipment has a safety factor of 5:1). The use of a motorized chain hoist between the overhead crane and the special lifting device further increased the probability of a load drop. The licensee lost focus of the risk mitigation measures included in the heavy load program and lifted the motor over RHR Trains A and B without isolating the RHR trains from the RCS. The nuclear safety concern (i.e., loss of reactor coolant inventory and residual heat removal capability) associated with the increased probability of a load drop could have been eliminated by performing the lift after the fuel had been transferred from the reactor vessel to the spent fuel pool, which occurred later in the outage.

In summary, recent nuclear industry heavy load drops resulting from failures of “below-the-hook” devices have involved cutting of synthetic slings and intermediate hoist component failure. The synthetic slings were cut through contact with inadequately protected load corners. The use of an intermediate hoist for heavy load lifts is inconsistent with the guidelines of NUREG-0612.

6. Crane Inspection, Testing, and Maintenance

Section 5.1.1 of NUREG-0612 states that the crane should be inspected, tested and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, "Overhead and Gantry Cranes." The crane has many components whose failure could reasonably result in the drop of a load, including the holding brake, the wire rope, and the load blocks. In addition, combinations of operator error and equipment failure have resulted in load drops or other damage.

Malfunctions of the holding brakes, the hoist motor and the associated controls have been identified as a contributor to crane operational events. These components are critical to the crane operator maintaining control of the load motion. Inspection, testing, and maintenance reduce the probability of uncontrolled load motion and ensure that protective devices function correctly to prevent a load drop.

Load drop events identified in Appendix A of NUREG-1774 include events where the wire rope of the crane was cut or failed due to the load block making contact with the upper block. This condition is known as "two-blocking." These two-blocking events are among the most common causes of load drops related to overhead crane failures at nuclear power plants. Inspection, testing, and maintenance affect the probability of these types of events because these events typically involve improper operation of the hoist upper limit switch and may involve improper operation of the hoist controls. A recent example of a hoist control problem occurred at Millstone Unit 3 when the crane malfunctioned: it continued to lift the load block although the controls were returned to the neutral, hold position. The missile shield lifting rig that was connected to the load block at the time experienced significant damage when fixed equipment interfered with the upward motion of the lift rig. Removing power to the crane halted upward motion, and the cause of the malfunction was identified as a stuck relay in the hoist controls. A brief description of this event was included in NRC Inspection Report 05000423/2004006 (ML042110368), dated July 29, 2004.

7. Crane Design

Section 5.1.1 of NUREG-0612 states that cranes should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, "Overhead and Gantry Cranes," and CMAA-70, "Specifications for Electric Overhead Traveling Cranes." If cranes are designed to these standards, then NRC has assurance that the crane can accommodate reasonable wear and degradation without a failure that could result in a load drop.

B. SUPPLEMENTAL GUIDELINES (SECTIONS 5.1.2–5.1.6 OF NUREG-0612)

Sections 5.1.2–5.1.5 of NUREG-0612 provide recommended guidelines to supplement the general guidelines for specific areas within a nuclear power plant. In general, these guidelines add a measure of defense-in-depth by either maintaining horizontal separation between the load and irradiated fuel or essential safe-shutdown equipment, improving the reliability of the load handing system by making it single-failure-proof, or demonstrating through analyses of potential load drops that the consequences are acceptable. The NRC staff considers the consequences acceptable when the criteria of Section 5.1 of NUREG-0612 are satisfied with respect to limited radiological releases, maintenance of a margin to criticality, limited damage to the spent fuel pool or reactor vessel, and maintenance of safe shutdown functions. Load drop analyses are not necessary when a single-failure-proof handling system is employed or horizontal separation is maintained by physical interlocks because the probability of the load imparting significant energy to irradiated fuel or essential safe-shutdown equipment as a result of a handling system failure is very small.

1. Consequence Evaluation of Load Drops

Appendix A to NUREG-0612 describes recommended assumptions and considerations for evaluating the consequences of postulated load drops. The general considerations include the following key assumptions: (1) the load is dropped in an orientation that causes the most severe consequences, (2) the analysis is based on an elastic-plastic curve that represents a true stress-strain relationship, and (3) all the energy of the drop is absorbed by the structures and equipment that are impacted. Appendix A also includes specific assumptions and considerations for analyses involving reactor vessel head drops, spent fuel cask drops, and spent fuel pool and reactor vessel margin to criticality.

As noted in the introduction, the survey of operating experience found that the calculational methodologies, assumptions, and predicted consequences of load drop events varied greatly from licensee to licensee for similar scenarios. Load drop calculations are used to determine lift height restrictions and locations where other measures to limit or prevent damage from postulated load drops, such as impact limiting devices, crane motion interlocks, or single-failure proof handling systems, are necessary. Varying degrees of conservatism in the assumptions and methodologies used in the analyses could be the cause of the observed disparity in load drop analysis results. However, NRC inspectors have identified recent issues involving the use of non-conservative assumptions and inadequate resolution of load drop analysis results indicating the potential for significant radiological consequences.

i. Non-conservative Assumptions and Methodologies

In NRC Inspection Report 05000282/2005004; 05000306/2005004 for Prairie Island Units 1 and 2 (ML052020420), dated July 21, 2005, NRC inspectors described the review of the revised reactor vessel head drop analysis in preparation for reactor vessel head replacement with an integrated head package weighing more than the original head. The analysis concluded that an accidental reactor head drop over irradiated fuel in an open reactor vessel would not adversely affect the functionality of the safety injection system. The methodology, design requirements, and acceptance limits used in the analysis were derived from the original head drop analysis. This analysis provided the basis for the maximum allowed lift elevation over irradiated fuel in an open reactor vessel, which was included in maintenance procedures for reactor vessel head removal and documented in the safety analysis report.

The inspector identified non-conservative and unjustified assumptions related to the evaluation methodology and acceptance criteria. The evaluation assumed a perfectly inelastic, plastic collision (i.e., the reactor vessel head and reactor vessel move in unison at the same velocity following impact). This assumption is non-conservative because it results in the minimum amount of energy being absorbed by the reactor vessel piping and support structures while still conforming to the principal of conservation of momentum. The licensee then assumed that the remainder of the energy from the drop was absorbed by plastic deformation at the interface between the vessel and the head, where the damage does not affect essential safety functions.

An independent licensee contractor modeled the postulated reactor vessel head drop using non-linear, time-history, and finite element analysis methods. The model was used to evaluate the effect of the impact on the reactor vessel and its supporting components in order to demonstrate that the safety injection system would remain capable of injecting water into the reactor vessel to remove decay heat. With reasonable weight and lift elevation restrictions, the calculation demonstrated that the reactor vessel components were structurally stable and total deformation would remain within limits that provide assurance of continued safety injection system functionality. However, the revised analysis reduced the maximum reactor vessel lift height from 10.8 meters (35.5 feet) to 8.2 meters (27.0 feet), confirming that the original assumptions and methodology were non-conservative.

ii. Incomplete Resolution of Load Drop Analysis Results

In a Safety Evaluation Report (SER) dated June 24, 2005 (ML051750678), the NRC staff described the resolution of a previously incomplete load drop analysis for Point Beach Nuclear Plant (PBNP), Unit 2. The licensee for PBNP performed the initial analysis of a reactor vessel head drop in response to a request from the NRC in a letter dated December 20, 1980, later identified as Generic Letter (GL) 80-113. In a letter dated November 22, 1982, Wisconsin Electric Power Company submitted to NRC the results of a reactor vessel head drop analysis. This letter stated that:

> The results of this analysis show that upon impact of the head drop the initial reactor vessel nozzle stresses are well within allowables. However, the loads imposed upon the reactor vessel supports caused by the impact of the head are greater than the critical buckling load of the support columns. These supports cannot be relied upon to absorb enough of the energy of impact to prevent severe damage to the safety injection lines attached to the reactor vessel or to the primary coolant loop piping.
>
> The results of the head drop analysis are presently being reviewed. This review is comprised of the following actions:
>
> (1) A review of the consequences of the head drop event for comparison with the guidelines of Section 5.1 of NUREG-0612, “Completion of Phase II of ‘Control of Heavy Loads at Nuclear Power Plants.’ ”
>
> (2) An identification of alternative measures which may be used to remove decay heat from the core should normal methods of residual heat removal (RHR) become inoperative.

(3) A determination of the probability of a head drop event based upon a lift frequency and current reactor operating history.

(4) A determination of any potential modifications which could be made to limit the probability of occurrences of a head drop event.

(5) A detailed review of the containment polar crane to determine areas of potential single failure that could be upgraded to provide increased reliability.

It is anticipated that the review process will be concluded within our originally proposed time frame for NUREG-0612 compliance, that is, January 1984.

However, it is unlikely that equipment modifications could be accomplished within this time frame. Should they be needed, such modifications would be completed as expeditiously as possible.

Although this analysis presented results that did not meet the acceptance criteria of Section 5.1 of NUREG-0612, the licensee identified potential courses of action that could reduce the probability of a reactor vessel head drop or mitigate the consequences of such an event. However, with the exception of actions 3 and 5, the licensee was not able to provide records showing that these actions were completed. As stated in GL 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-related Equipment”:

> ... the [NRC] staff is concerned that other licensees may believe that their heavy load operations are in compliance with the regulations because they have completed Phase I of the GL of December 22, 1980, and the closeout of Phase II by GL 85-11. GL 85-11 did not relieve licensees of their responsibility under 10 CFR [Title 10 of the *Code of Federal Regulations*] 50.59 to evaluate new activities with respect to the SAR [safety analysis report] and the Technical Specifications to determine whether the activity involves an unreviewed safety question or a change in the Technical Specifications. In addition GL 85-11 concluded that the risks associated with damage to safety-related systems are relatively small because (1) nearly all load paths avoid this equipment, (2) most equipment is protected by an intervening floor, (3) there is redundancy of components, and (4) crane failure probability is generally independent of safety-related systems. As is demonstrated by Oyster Creek’s proposed activities [movement of a 100-ton spent fuel cask over equipment essential for safe shutdown that was not adequately protected by the intervening floor], this conclusion may not always be valid.

Since the 1982 Point Beach reactor vessel head drop analysis was submitted to NRC based on a request from the NRC staff, 10 CFR 50.71(e) required that the results of the evaluation be incorporated into the safety analysis report. In order to incorporate the 1982 reactor vessel head drop analysis into the safety analysis report, the licensee completed a 10 CFR 50.59, “Changes, tests, and experiments,” review in April 2005. This review concluded that the proposed change to the safety analysis report required prior NRC approval in accordance with the requirements of 10 CFR 50.59(c)(2)(v), and the licensee submitted a license amendment request (LAR) in accordance with the requirements of 10 CFR 50.90.

The 1982 reactor vessel head drop analysis was limited to elastic behavior of the structures, piping, and components that are impacted. The licensee, with support from Sargent & Lundy (S&L) and Westinghouse, determined that inelastic structure and piping behaviors would absorb significant energy such that there is reasonable assurance that the pressure boundary integrity of the reactor coolant system piping for PBNP Unit 2 would be maintained in the event of a postulated reactor vessel head drop. In addition, the licensee assessed and found acceptable the reliability of the handling system, the availability of other methods to remove decay heat, and the potential radiological consequences of a postulated reactor vessel head drop.

2. Upgrade of Existing Cranes to Single-Failure-Proof (NUREG-0554 and Section 5.1.6 and Appendix C to NUREG-0612)

In evaluating changes to heavy load handling programs, particularly in relation to initiating dry storage of irradiated fuel, many licensees elect to upgrade the overhead crane. The NRC staff has approved use of a single-failure-proof crane designed to the guidelines of NUREG-0554, in conjunction with implementation of the general guidelines of Section 5.1.1 of NUREG-0612, as an acceptable method for maintaining safety when handling heavy loads over spent fuel storage areas and essential safe-shutdown equipment. Section 5.1.6 and Appendix C to NUREG-0612 describe alternative methods of complying with NUREG-0554 guidelines for existing cranes.

i. Evaluation of Existing Crane Bridge

In upgrading an existing crane to single-failure-proof standards, the existing crane bridge is often reused, while the crane trolley and hoist components are typically replaced in their entirety. The existing crane bridge was often constructed to crane manufacturer standards used for all industrial crane applications. The stricter criteria of NUREG-0554 provide assurance that the crane will stop and hold the load under all credible conditions.

These criteria include cold proof testing of the crane bridge when the material properties of the bridge are not known. Cold proof testing provides assurance that the material of the crane would not be subject to brittle failure by verifying the structural integrity of the crane during a test at 125% of rated load when the metal temperature of the crane is at a temperature below the operating temperature of the crane. Section 2.4 of NUREG-0554 specifies nondestructive examination following the cold proof test of accessible welds whose failure could cause the load to drop. Appendix C to NUREG-0612 provides an alternative to cold proof testing. When an existing crane is operated at temperatures more than 33 ˚C (60 ˚F) above the nil-ductility transition temperature for the structural steels used for the crane structure, the cold proof test may be omitted.

Section 2.8 of NUREG-0554 specifies that critical welds (i.e., those welds whose failure could result in a load drop) should be completed at controlled temperatures and heat-treated after welding to relieve residual stress. For existing cranes, Appendix C to NUREG-0612 permits non-destructive examination of the accessible critical welds as a substitute for heat treatment when the welds may not have been heat treated in accordance with AWS D1.1, "Structural Welding Code." Many existing cranes lack the records necessary to demonstrate the welds have been acceptably heat treated. Appropriate non-destructive examination methods would be capable of detecting defects with the potential to threaten the integrity of the bridge structure that would credibly result from inadequate heat treatment of the weld and surrounding heat affected zone. Often, magnetic particle testing is appropriate for the configuration, but other methods such as radiography or ultrasonic testing may be necessary for certain weld configurations. Since the crane bridge and trolley wheels must stay on their respective runway rails to ensure that neither the load nor the crane itself drops, the scope of the non-destructive examination should begin with the truck structure that supports and aligns the wheels on the rails. Other important welds include those that align the wheel trucks relative to the bridge girders and critical welds in the bridge girders themselves.

Recent operating experience includes an example where inadequate welds on the crane bridge adversely affected crane operation. In NRC Inspection Report 07200034/2004001 (ML042740167), dated September 10, 2004, inspectors described the auxiliary building crane problems encountered during initial cask loading at Sequoyah Nuclear Power Plant in June 2004, the cause of the crane problems, and subsequent crane repairs and testing to correct the problems.

During the first loaded spent fuel cask movement, the auxiliary building crane bridge drive motor became overloaded and tripped on two occasions. Through subsequent inspections, the licensee discovered cracks in welds between the bottom flanges of all four crane-trucks adjacent to the seismic restraint, which extended up to 30 centimeters (12 inches) into the base metal of the I-beam web. A total of more than 20 cracks were identified in the four trucks using visual and magnetic particle examinations. The cracks allowed the wheels to move upward relative to the bottom flange of the trucks, which created interference between the bridge rail anchors and the crane truck seismic restraints during crane motion.

The licensee performed weld and base metal repairs by excavating the weld and base metal indications to sound metal and performing weld repairs. The licensee radiographed all weld and base metal repairs to ensure that all cracks and rejectable weld indications had been removed. Crane testing was performed after the repairs were completed, which consisted of both a 100% full range of motion test and a 125% load test. The licensee performed visual inspections after each load test to examine the weld and base metal areas for cracks in the entire crane structure, including the areas of modification.

The inspectors found that the crane had a long history of broken bridge rail anchor bolts dating to 1985, and that the cause had not been correctly identified. Although the area where the cracks developed was difficult to access, the welds where the cracking initiated were not inaccessible. The periodic complete inspections of the crane performed in accordance with paragraph 2-2.1.3 of ANSI B30.2, "Overhead and Gantry Cranes," did not include the structure containing the cracked areas due to concerns regarding accessibility. In addition, the licensee performed visual inspections of critical welds during the crane upgrade to single failure proof status. The welds inspected were identified as critical welds by the crane vendor and included horizontal welds between girder top or bottom plates and web plates, and trolley load girder. The welds in the crane trucks were not identified as critical welds by the crane vendor and were therefore not inspected.

All the welds in the load path, including welds in the crane trucks, that are used to carry, transfer, or retain the critical loads and prevent a load drop are within the scope of the welds specified within NUREG-0554 as critical. Section 2.6 of NUREG-0554 specifies that those welds be subject to nondestructive examination, and Section 10 of NUREG-0554 specifies that a quality assurance program addressing all recommendations of NUREG-0554 should be implemented. Therefore, nondestructive examination should be consistent with the quality assurance program established for the single-failure-proof handling system.

ii. Seismic Evaluation of Crane Structures

Section 2.5 of NUREG-0554 states that cranes should be designed to retain control of and hold the load with the bridge and trolley on their respective runways and their wheels prevented from leaving the tracks during a seismic event comparable to the safe shutdown earthquake. The evaluations necessary to demonstrate this capability should be consistent with the plant licensing basis for seismic evaluations. Section 2.5 also states that the crane should be designed and constructed to satisfy regulatory position 2 of Regulatory Guide 1.29, “Seismic Design Classification.” This capability provides assurance that neither the crane structures nor the load would become a missile that could damage irradiated fuel or safety-related equipment. Where assumptions specific to the crane configuration are necessary for evaluating the structural response to a seismic event, the assumptions should result in a realistic or conservative modeling of the crane seismic response.

Section 2.5 of NUREG-0554 states that seismically induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge. Seismically induced pendulum and swinging loads may have a significant horizontal component when the structure supporting the crane trolley either experiences a sudden, large displacement or is displaced small distances at a frequency near the natural frequency of the pendulum formed by the hoist rope, lifting device, and load. However, the additional vertical load resulting from swinging of the load is negligible because the arc of travel, and the associated tangential velocity, would be small.

On May 16, 2003, the staff issued Amendment 251 to the operating license for the Duane Arnold Energy Center (ML0309900410), which accepted the reactor building crane as single-failure-proof. After the issuance of Amendment No. 251, the NRC staff determined that the SER accompanying the amendment should be supplemented to ensure proper documentation of the judgement exercised by the licensee, and the basis for the staff's agreement with that judgement, regarding seismically induced pendulum and swinging loads. Therefore, the NRC staff requested the licensee to document the basis for its conclusion regarding the effects from the horizontal seismic excitation on the swinging load. In response to this NRC staff request, the licensee stated in a letter dated January 7, 2004 (ML0401505271), that:

> The fundamental building frequency of the reactor building structure is much higher than that of the crane/load system. The response spectra for the building peak at a period of less than 1/3 second. An informal review of the crane/load system indicates that the shortest expected period would be greater than 3 seconds. Therefore, the horizontal seismic forces exerted on the suspended load would have no appreciable effect on the crane or the building structure.

In a supplemental safety evaluation for the Duane Arnold Energy Center dated January 30, 2004, the staff stated (ML0402201870):

> The NRC staff has verified in an independent analysis that, as the licensee states above, the fundamental period of the crane/load system is indeed much larger than that for the reactor building. Furthermore, from its examination of DAEC's safe shutdown earthquake (SSE) spectra, the NRC staff finds that the horizontal seismic excitation above a natural period of 1 second is insignificant (less than 0.1g). Therefore, the largest dynamic responses of the reactor building, crane, and supporting structure resulting from the input seismic ground motion would be at periods that are much lower than the natural period for the swinging load, implying that the load would not be excited by the building motion. Of further concern is the possibility that the motion of the reactor building, crane, and supporting structure during an earthquake is significant enough that the load does not remain directly below the crane. However, the largest spectral displacement from the DAEC SSE is only 4 inches at a natural period of 10 seconds. In addition, structural amplification of these long period ground motions at the elevation of the crane at 200 feet above the foundation is not likely to be significant. A natural period of 10 seconds corresponds to an 80-foot long pendulum (a possible configuration at a boiling-water reactor facility), and a 4-inch displacement at this length would add a negligible amount to the horizontal forces acting on the crane. Thus, the NRC staff concludes that the swinging load effects on the crane are negligible and that the licensee's assumption that the lifted load and lower load block are decoupled from the bridge and trolley with respect to horizontal earthquake accelerations is appropriate. On this basis, the licensee's conformance with the provisions in Section 2.5 of NUREG-0554 in the seismic analyses is acceptable.

Therefore, under certain conditions, seismically induced pendulum loads may contribute to the seismic loads acting on the crane structure. Considerations in assessing the significance of the lifted load contribution to the overall seismic load on the crane include: the dynamic response of the structure supporting the crane at the elevation of the crane; the relationship of the maximum displacement of the structure relative to the distance the load is suspended below the crane; and the relationship of the dominant structural response frequency to the natural frequency of the suspended load.

C. CHANGES TO THE HEAVY LOAD HANDLING PROGRAM

The heavy load handling program may require changes to allow for new operations, such as beginning dry cask storage of spent fuel, or to address other movements of heavy loads that are outside the bounds of the existing heavy load handling program, such as movements associated with facility modifications and replacement of certain heavy components. Licensees must review changes in accordance with the requirements of 10 CFR 50.59, "Changes, tests and experiments." Licensees may determine that certain changes require a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit." Additionally, licensees may determine that an update to the safety analysis report to reflect the change is necessary pursuant to 10 CFR 50.71, "Maintenance of records, making of reports."

Cranes are predominantly used in activities that could be classified as maintenance activities. Many of the elements of the heavy loads handling programs at each nuclear power plant are measures that manage the increase in risk that results from heavy load handling activities associated with maintenance. Therefore conformance with the heavy loads handling program complies with the requirements of 10 CFR 50.65(a)(4) with regard to managing the risk associated with heavy load movements in support of maintenance activities.

When a licensee identifies a new heavy load handling evolution that is not bounded by previously evaluated load movements, the licensee should evaluate the change in accordance with 10 CFR 50.59 to determine the need for a license amendment. Consistent with NRC guidelines included in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," and Nuclear Energy Institute (NEI) 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," which the Regulatory Guide endorses, licensees may consider a change to a single-failure-proof load handling system that fully conforms with the guidelines of Section 5.1.6 of NUREG-0612, an NRC approved method of safely handling heavy loads.

CONCLUSION

Heavy load handling activities pose a safety risk in areas of nuclear power plants where load drops could damage irradiated fuel or equipment necessary for safe shutdown. NRC developed the guidelines in Section 5.1 of NUREG-0612 to reduce the frequency of heavy load drops at nuclear power plants and provide a measure of defense-in-depth. The defense-in-depth measures provide assurance that the probability of a load handling event that damages irradiated fuel or safe shutdown equipment is acceptably small. Licensees incorporated many of these recommendations into their site-specific heavy load handling programs. However, operating experience and inspection findings related to heavy loads handling indicate that additional clarification and reemphasis of heavy load handling guidelines may reduce the frequency of load handling events. Improving human performance in rigging activities is an area of particular concern based on recent operating experience.