September 5, 2008

Mr. Thomas C. Houghton, Director
Strategic Regulatory Programs, Nuclear Generation Division
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006-3708

SUBJECT: INDUSTRY INITIATIVE ON CONTROL OF HEAVY LOADS

Dear Mr. Houghton:

The Nuclear Regulatory Commission (NRC) staff has completed its evaluation of guidance the Nuclear Energy Institute (NEI) developed for the implementation of the industry initiative on control of heavy loads. Attached to this letter for your information is the NRC staff safety evaluation addressing NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0.

The attached safety evaluation constitutes formal NRC staff endorsement of the methods in NEI 08-05 for implementing the specified application, with the exception that the NRC staff considers the acceptance criteria of Appendix F to the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, (rather than the industry proposed strain-based criteria) appropriate for evaluation of coolant retaining component performance following a postulated head drop. The approval of the NRC staff may be important for licensees that conclude use of an NRC approved method would allow implementation of the initiative without a license amendment, pursuant to the requirements of 10 CFR 50.59. Application of the NEI guidelines for other purposes (e.g., application for 10 CFR 50.65(a)(4) risk management evaluations) may not require NRC staff approval. The staff intends to issue a Regulatory Issue Summary to inform industry of the endorsement of NEI 08-05 through the safety evaluation.

The NRC staff has concluded that NRC approved acceptance criteria for evaluation of coolant retaining component performance (i.e., Appendix F to the ASME B&PV Code, Section III, Division 1) are appropriate considering the broad acceptance of the ASME Code for evaluation of impact loading and the uncertainty in the analysis. We acknowledge industry concerns that these criteria are too conservative for the reactor vessel head drop analysis application. However, the NRC staff has identified alternative approaches (i.e., single failure proof crane equivalence or benchmarking of the analysis to reduce analytical uncertainty) that could be used if application of the NRC staff approved acceptance criteria is not workable.

As I stated in my letter dated May 29, 2008, the NRC staff recognizes that facility owners that ultimately intend to develop more detailed load drop analyses using energy balance methods or enhance the reliability of the handling system to single failure proof crane equivalence may need additional time to complete such efforts. Therefore, development of an interim analysis may be necessary and is an acceptable approach. Licensees may perform these interim analyses for reactor vessel head lifts over flooded refueling cavities that conform to the proposed analysis guidelines contained in NEI 08-05, with the exception that licensee need not evaluate the performance of the reactor vessel supports for reactor vessel head drops over flooded refueling cavities.

The NRC staff appreciates your development of the guidelines contained in NEI 08-05 and the continued industry interest in implementation of your initiative on heavy load handling. Please contact Mr. Steven Jones of the NRC staff at (301) 415-2712 to further discuss any issues.

Sincerely,

/RA/ Jared Wermiel for

William H. Ruland, Director
Division of Safety Systems
Office of Nuclear Reactor Regulation

PROJ: 689

cc: See Next Page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO NUCLEAR ENERGY INSTITUTE (NEI) 08-05, REVISION 0

INDUSTRY INITITATIVE ON CONTROL OF HEAVY LOADS

1.0 INTRODUCTION

By letter dated July 28, 2008, the Nuclear Energy Institute (NEI), transmitted NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0, July 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082180666). An industry task force sponsored by NEI developed the guideline document for use in implementing the industry initiative on control of heavy loads at nuclear power plants.

NEI described the initiative in a letter dated September 14, 2007(ADAMS Accession No. ML072670127), and stated that the initiative was intended to address a lack of consistency in plant licensing bases that the industry had identified related to control of heavy loads. The formal industry initiative included the following elements to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices:

- For all heavy load lifts, adequately implement commitments to safe load paths, load handling procedures, training of crane operators, use of special lifting devices, use of slings, crane design, and inspection, testing, and maintenance of the crane.

- For reactor vessel head lifts and spent fuel cask lifts over the spent fuel pool, use a single failure proof crane or establish a load drop analysis (generic or plant-specific) that bounds your planned lifts with respect to load weight, load height, and medium present under the load. Procedures for moving these loads reflect the safety basis. Load drop analyses can be based on realistic (i.e. best estimate) calculations.

- Ensure maintenance rule (a)(4) administrative controls include the movement of heavy loads as a configuration management activity.

- In the next safety analysis report update, provide a summary description of the basis for conducting safe heavy load movements, including commitments to safe load paths, load handling procedures, training of crane operators, use of special lifting devices, use of slings, crane design, and inspection, testing, and maintenance of the crane. If the safety basis includes reliance on a load drop analysis, then that fact should be included in the summary description within the safety analysis report.

- If load drop analyses are used, reflect restrictions on load height, load weight, and medium present under the load in plant procedures.

To support the initiative, NEI 08-05 includes guidelines for the following activities:

Enclosure

- Managing the risk associated with maintenance involving movement of heavy loads

- Performing consequence analyses for postulated reactor vessel head drops

- Establishing single-failure-proof equivalence for handling systems when used for reactor vessel head lifts

- Updating the description of heavy load handling programs in the safety analysis report

In a letter from William H. Ruland, Director, Division of Safety Systems, Office of Nuclear Reactor Regulation, U.S. NRC, to Thomas C. Houghton, Director, Strategic Regulatory Programs, Nuclear Generation Division, NEI, dated May 16, 2008 (ADAMS Accession No. ML081330440), the NRC staff presented its position on preliminary industry guidance for performing consequence analyses for postulated reactor vessel head drops and establishing single-failure-proof equivalence for reactor vessel head handling systems (ADAMS Accession Nos. ML081300340 and ML081300045, respectively). A separate letter from William H. Ruland to Thomas C. Houghton dated May 27, 2008 (ADAMS Accession No. ML081410597), the NRC staff clarified the approaches the staff considers acceptable for interim reactor vessel head drop analyses over flooded refueling cavities until more detailed load drop analyses using energy balance methods or enhancements to the reliability of the handling system are complete. The NRC staff is preparing this safety evaluation to more fully document the regulatory basis for endorsing, with exceptions, the industry guidance contained in NEI-08-05, Revision 0.

2.0 REGULATORY EVALUATION

The overhead heavy load handling systems at nuclear power plants support activities essential to the continued operation of nuclear power plants, such as refueling of the reactor vessel and, as the irradiated fuel inventory approaches the spent fuel pool capacity, transfer of irradiated fuel from the spent fuel pool in casks. However, some of the heavy loads handled by the overhead handling system could, if dropped, develop sufficient kinetic energy to damage structures, systems, or components that perform essential safe shutdown functions. The spent fuel pool structure and the reactor vessel combined with portions of the reactor coolant system perform a pressure boundary function essential for the removal of residual heat from irradiated fuel. The configuration of nuclear power plants results in reactor vessel heads and spent fuel casks commonly being lifted to substantial heights above the reactor vessel flange and spent fuel pool floor, respectively. Therefore, safe handling of reactor vessel heads and irradiated fuel casks is important to the essential safe shutdown function of residual heat removal.

General Design Criterion (GDC) 4, “Environmental and Dynamic Effects Design Bases,” of Appendix A to Title 10 of the Code of Federal Regulations (10 CFR) Part 50 specifies, in part, that structures, systems, and components (SSCs) important to safety shall be appropriately protected against dynamic effects, including the effects of missiles, that may result from equipment failures. Handling system failures could result in missiles that could affect SSCs important to safety.

In NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” issued July 1980 (ADAMS Accession No. ML070250180), the NRC staff provided regulatory guidelines for control of heavy load lifts to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Section 5.1.1 of NUREG-0612 provides guidelines addressing the following items to reduce the likelihood that heavy load handling would affect important-to-safety SSCs: development of safe load paths; development of procedures for load-handling operations; training of crane operators; and design, testing, inspection, and maintenance of cranes and lifting devices. The guidelines in Sections 5.1.2 through 5.1.6 address alternatives that either further reduce the likelihood that heavy load handling would affect important-to-safety SSCs or establish bounds on the operation of the heavy load handling system to ensure the consequences of a handling system failure would be acceptable. These alternatives include performance of load drop consequence analyses or use of a single-failure-proof crane to improve reliability through increased factors of safety and through redundancy or duality in certain active components. The NRC staff provided guidelines for performance of load drop consequence analyses in Appendix A to NUREG-0612, “Analyses of Postulated Load Drops.” Guidelines for the design of single-failure-proof cranes are included in NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” and the NRC staff provided guidelines for modifications to enhance the reliability of existing cranes in Appendix C to NUREG-0612, “Modification of Existing Cranes.” In a letter dated December 22, 1980, later identified as Generic Letter (GL) 80-113 (ADAMS Accession No. ML071080219), as modified by GL 81-07 (ADAMS Accession No. ML031080524), “Control of Heavy Loads,” dated February 3, 1981, the NRC staff requested that all licensees describe how they satisfied the guidelines of NUREG-0612 at their facility and what additional modifications would be necessary to fully satisfy these guidelines. The NRC staff divided this request into two phases (Phase I and Phase II) for implementation by licensees. The NRC staff requested Phase I responses within 6 months that addressed the guidelines in Section 5.1.1 of NUREG-0612. The NRC staff requested Phase II responses within 9 months that principally addressed Sections 5.1.2 through 5.1.6 of NUREG-0612. The Phase I and Phase II responses formed the bases for heavy load handling programs at nuclear power plants.

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

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

The staff closeout of the Phase II reviews in this manner resulted in a lack of consistency in plant licensing bases with respect to control of heavy loads. Some licensees had installed single-failure-proof cranes, and some other licensees had developed load drop consequence analyses. However, the appropriate regulatory treatment of licensing basis information related to these activities was unclear.

Implementation of the voluntary industry initiative on control of heavy loads may involve changes in the facility as described in the safety analysis report or changes in the procedures as described in the safety analysis report. Licensees must review these changes in accordance with the requirements of 10 CFR 50.59, “Changes, Tests and Experiments.” Industry guidance endorsed by this safety evaluation for a specific application may be considered as a method approved by the NRC for that application pursuant to 10 CFR 50.59(a)(2). Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, ‘Changes, Tests, And Experiments’,” endorsed Revision 1 of NEI 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” dated November 2000 (ADAMS Accession No. ML003771157), as providing methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59.

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.” Regulatory Guide 1.181, “Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e),” endorsed Revision 1 of NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports,” dated June 1999, (ADAMS Accession No. ML003779028), as providing methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.71(e).

Heavy load handling is often a part of 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 could result from heavy load handling associated with maintenance. Pursuant to the requirements of 10 CFR 50.65(a)(4), licensees shall assess and manage the increase in risk that may result from proposed maintenance activities. Therefore the heavy loads handling program is associated 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. Regulatory Guide 1.182, “Assessing and Managing Risk Before Maintenance Activities At Nuclear Power Plants,” endorsed Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, (ADAMS Accession No. ML003704489), as providing methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.65(a)(4).

3.0 EVALUATION OF INDUSTRY INITITATIVE ON CONTROL OF HEAVY LOADS

An industry task force on heavy loads established by NEI developed the guidelines included in NEI 08-05 for implementation of the initiative. Section 1 of NEI 08-05 provides guidance on maintenance rule (i.e., 10 CFR 50.65(a)(4)) administrative controls. Section 2 of NEI 08-05 provides guidance on reactor vessel load drop analysis, and Section 3 provides guidance on reactor head lift single failure proof crane equivalence. The NRC sponsored several public meetings with NEI representatives to discuss the content of Sections 2 and 3. (Summaries of meetings held on December 13, 2007, February 1, 2008, April 8, 2008, and April 17, 2008, are available at ADAMS Accession Nos. ML080100159, ML080570529, ML081050266, and ML081300823, respectively.) Section 4 provides guidance on updating the safety analysis report.

3.1 Managing the Risk of Heavy Load Handling For Maintenance

Consistent with the NRC staff position described in RG 1.182, Section 1 of NEI 08-05 notes that NUMARC 93-01, Revision 3, Section 11 provides general guidance for assessment and management of risk due to maintenance activities under 10 CFR 50.65(a)(4). The guideline provides additional guidance, assessment considerations, and risk management actions specific to maintenance activities involving handling of heavy loads.

The additional guidance in Section 1 of NEI 08-05 discusses the scope and method of evaluating risk. Section 9.1.5, Revision 1, of the NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, provides guidance that defines a heavy load as a load weighing more than one fuel assembly and its handling device. The weight of the lightest fuel assemblies (i.e., those used in boiling-water reactors) and their associated handling device is approximately equal to the 1000-pound criterion proposed in NEI 08-05. Therefore, a lower bound of approximately 1000 pounds is consistent with NRC staff positions. The additional guidance also states that a quantitative risk assessment is not necessary, which is consistent with the guidelines of Section 11 of NUMARC 93-01.

The considerations discussed in Section 1 of NEI 08-05 include whether a train or other equipment under the load path is protected, whether the handling system is single failure proof, and a general assumption that the safety function would be impaired by a potential load drop. With the clarification that redundant trains of equipment essential for safe shutdown should be considered when a potential load drop in a single location could affect both trains, the NRC staff finds the considerations appropriate because they are generally consistent with the guidelines of Section 9.1.5 of NUREG-0800.

The risk management actions described in Section 1 of NEI 08-05 include revising the load path to avoid essential equipment, providing compensatory measures or backup equipment to enhance the redundancy available for safety function performance, and administrative controls related to operational awareness and managerial approval. For application under the industry initiative on control of heavy loads, the risk management actions are adequate to ensure loads are handled in a manner that minimizes lifts over essential equipment. When lifts over essential equipment are necessary, use of a single failure proof handling system, use of compensatory measures to reduce the probability that a load drop could affect the essential safety function, or provision of backup equipment that can perform the essential function are acceptable means of managing the risk associated with potential damage to essential equipment.

3.2 Reactor Vessel Head Load Drop and Consequence Analysis

The guidance contained in Section 2 of NEI 08-05, "Industry Criteria for Reactor Vessel Head Load Drop and Consequence Analysis," includes: (1) a comparison of NUREG-0612 guidelines for analyses of postulated reactor vessel head drops and the industry initiative guidelines, (2) general guidelines for the analysis, (3) selection of material properties, (4) analytical modeling requirements, and (5) acceptance criteria when evaluating the effects of postulated heavy load drops. The industry's purpose in developing these guidelines on reactor vessel head drop analyses was to provide consistency in plant licensing basis information used to demonstrate that, after a postulated reactor vessel head drop accident, the core remains covered with coolant and sufficient cooling is available. However, the industry chose not to endorse a specific methodology.

Comparison with NUREG-0612 Guidelines

Table 1 of Section 2 of NEI 08-05 provides a comparison between the guidelines included in Section 5.1 and Appendix A of NUREG-0612 and the industry guidelines to be used for reactor vessel load drop evaluations as part of the industry initiative. The industry guideline limits the scope of evaluation to cases that, based on previous evaluations, have been determined to represent worst case conditions. Specifically, the scope of the evaluation is limited to demonstrating that the core remains covered with coolant and sufficient cooling is available. Also the assumed configuration of the postulated drop is limited to a concentric drop of the reactor head from the highest elevation permitted by procedures while the center of gravity of the head is within the outer radius of the reactor vessel flange.

The NRC staff considers the scope of the evaluation acceptable. The reactor vessel head itself cannot credibly contact the fuel in the vessel directly. Previous evaluations have indicated that the consequences of impacts between the upper vessel internals and the fuel were not significant with respect to public health and safety. Consequently, the appropriate focus is demonstrating that, following postulated reactor vessel drops, the core remains covered with coolant and sufficient cooling is available.

The staff also found the assumed configuration of components for the load drop analysis acceptable. Operating experience demonstrates that special lifting devices are extremely reliable in holding the load once the lift has commenced (i.e., lifting device problems tend to be self-revealing at an early stage in the lift sequence). Credible failures in the handling system above the special lifting device would result in the head dropping in a flat configuration because the failure would occur at a point directly above the center of gravity of the head (i.e., no significant rotational moment to tip the head would result from the failure). Also, concentric flat drops are limiting for many plant configurations because any one of several symmetrically spaced safety injection lines would provide adequate coolant to maintain the core covered and uneven distribution of impact energy around the vessel would increase the likelihood of one line surviving the impact. Drops where the center of gravity of the head was outside the outer radius of the vessel flange would be likely to transfer substantial energy to structures other than the reactor vessel and, therefore, would not be limiting relative to maintaining the fuel covered with coolant. Drops from elevations above those allowed by procedures need not be considered because such postulated drops would require both a human performance issue in failing to control the height of the reactor vessel head and an additional mechanical failure of a crane component or multiple component failures. The assumed weight of the reactor vessel head includes the lifting device, but may exclude the weight of the lower load block because the time difference
during a postulated drop between the head impact with the vessel and the load block impact with the head decouples the two events.

<u>General Load Drop Analysis Considerations</u>

The modeling requirements and acceptance criteria contain guidelines that apply to classical component models using closed form solutions, finite element analyses, hybrid analyses, and comparative analyses where the individual parameters for a facility are compared to those used in a previously analyzed similar configuration to demonstrate acceptable outcomes. Where existing analyses have received NRC staff approval (e.g., through a safety evaluation), no further analysis is necessary. If existing analyses have not been approved by the NRC staff, the industry guidelines state that the licensee may compare its previous analysis to these guidelines to determine if more analysis is needed.

NEI updated the guidance document in response to NRC staff concerns discussed during the public meetings. These concerns involved the process used to establish material properties and the adequacy of the proposed strain acceptance criteria. These issues are discussed in more detail below.

Material Properties

The draft industry guidelines specify that minimum code or specification values for material strength or actual or representative test data (e.g., certified material test reports (CMTRs)) may be used to establish material properties. Stress-strain curves may be developed from test data or by modifying true stress-strain curves for similar material to match key material properties.

The staff was concerned that the CMTRs may provide unrealistically high values for material properties without suitable controls on the quality and diversity of the information. In response, NEI modified the guidance to specify that the resultant data account for uncertainties caused by variations in properties throughout the material. The guidance also specifies that, where multiple test results are available, minimum values for both stress and strain be used. The staff finds application of this revised guidance acceptable for establishment of material properties.

Analytical Methods

The industry guidelines specify considerations for finite element modeling including: element (mesh) sizing; modeling of the head; adequate representation of post-buckling, necking, and other material instabilities; and contact damping coefficients for components that remain elastic. The industry guidelines also include some more general modeling considerations related to both the classical and finite element analysis methods. The staff found these considerations and specific assumed values for certain effects not within the scope of the respective models acceptable.

Component Acceptance Criteria

The industry guideline provides acceptance criteria for equivalent force (stress-based) evaluations and strain-based evaluations. The industry specified acceptance criteria based on the following standards for equivalent force evaluations: American Society of Mechanical Engineers' (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Appendix F and, for concrete structures, American Concrete Institute (ACI) 349, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary." These standards are consistent with standards specified for evaluation of safety-related structures for limiting, low-frequency events and are acceptable.

The industry-proposed acceptance criteria for strain-based evaluations are more complex. NEI provided its technical justification for the proposed strain approach as an attachment to its July 28, 2008, letter. In that attachment, NEI opined that the ASME Code does not specifically address the physics associated with a head drop event. However, the NRC staff considers the head drop event to be a classic impact loading event. Paragraph NB-3111 of the ASME B&PV Code, Section III, specifies loads that must be considered in the evaluation of ASME Class 1 components (reactor vessel). Paragraph NB-3111(b) specifies impact loads as a loading condition that must be considered. Therefore, the ASME Code addresses impact loads such as the head drop.

The NRC staff considers the ASME B&PV Code, Section III, Division 1, Appendix F acceptance criteria for limiting events (i.e., Service Level D) acceptable for all analytical methods proposed in the draft guidance. For energy balance evaluations using the large-displacement finite element methods described in the guidance, the NRC staff considers the criteria applied to pipe whip restraint evaluations (i.e., one-half of ultimate strain) acceptable for application to component support evaluations.

In the guidance, industry has proposed more relaxed strain-based criteria for application to large-displacement finite element evaluations of coolant retaining components and component supports. The criteria would allow large inelastic deformations in the components and supports. The NRC staff concluded that these acceptance criteria do not provide sufficient margin for analysis uncertainty and, thus, the model could under-predict the deformations resulting from the head drop. Previous attempts to analyze components subject to large, dynamic, and inelastic deformations have shown that the analyses can significantly under-predict test results if the analysis model has not been properly benchmarked against actual tests. The staff may consider acceptance of the proposed relaxed strain-based criteria, if such criteria is developed in conjunction with accurate benchmarking to large displacement tests of similar material in similar configuration. Under such an approach, the proposed relaxed strain-based acceptance criteria may be acceptable as a result of the reduced uncertainty provided by the benchmarking.

Maximum Displacement

The guidelines specify that the maximum vertical deflection of the reactor vessel following the initial reactor head impact be limited to a deflection that would allow attached piping to continue supplying coolant to the reactor vessel. A supporting component or concrete foundation may fail the applicable component acceptance criteria, but the failure must be displacement limited and the reactor vessel attached piping must be capable of withstanding the additional displacement. This criterion is consistent with the fundamental acceptance criterion of maintaining the core covered with coolant and providing adequate cooling. Therefore, this criterion is acceptable.

Parametric Evaluations

The guideline document identifies considerations for performing parametric evaluations at plants with sufficiently similar reactor vessel and support configurations. The essential parameters for consideration are the height and weight of the reactor vessel head, the material properties of the reactor vessel and supporting structures, and the configuration and dimensions of the vessel and supporting structures. The parameter values used in the base analysis need not envelope each of the parameter values for the comparison analysis, but the guidelines state that the analyst should ensure that the effect of the individual parameter differences is properly weighted. The NRC staff finds the parameters identified for consideration in the parametric analysis adequate to ensure the comparison is accurate.

3.3 Single Failure Proof Crane Equivalence for Reactor Vessel Head Lifts

The industry initiative on Control of Heavy Loads provides the alternative of using a “single failure proof crane” for reactor vessel head lifts and spent fuel cask lifts over the spent fuel pool. Section 3 of NEI 08-05, “Reactor Head Lift Single Failure Proof Crane Equivalence,” provides industry guidance for establishing single failure proof equivalence for cranes for the limited purpose of lifting the reactor vessel head. This guidance does not apply to the lifting or movement of spent fuel casks over the spent fuel pool. It also does not apply to new cranes being ordered for new reactor construction.

In GL 80-113, the NRC staff provided the methodology for licensees to establish single failure proof handling systems. Licensees electing to establish a single failure proof handling system were requested to provide point-by-point comparison of the subject crane to the guidelines in NUREG 0554. Some licensees submitted responses that included this comparison. However, for many facilities, the safety benefit to fully upgrade the crane was insufficient to justify the expense of fully upgrading the crane to meet NUREG-0554 guidelines. In an attachment to GL 85-11, the NRC staff recognized this condition.

Section 3 of NEI 08-05 is based on Appendix C of NUREG-0612, which included a summary of design and hardware features specified in NUREG-0554 guidelines for single failure proof cranes. In Section 3 of NEI 08-05, the industry identified other measures intended to provide equivalent protection against load drops for some of the design and hardware features. Thus, the industry guideline provides a list of essential design and hardware modifications necessary to protect against significant causes of load drops and identifies other measures that provide equivalent protection at lower expense.

Section 3 of NEI 08-05 identifies the following design and hardware features as essential:

- master switches with spring return to off feature
- cab mounted emergency stop button
- two holding brakes
- two upper limit switches (second upper limit shall be a power disconnect)
- overspeed sensor/circuit

Measures to demonstrate equivalent protection for other NUREG-0554 specified design and hardware features include the following:

- Verification that the frequency of the safe shutdown earthquake (SSE) acceleration occurring with the reactor vessel head suspended over the reactor vessel, based on return period of SSE and the time the load is over the reactor vessel, is <1E-7 per year.
- Verification that the crane meets the allowable stresses for the bridge, end trucks, and trolley structural components specified in the Crane Manufacturers Association of America (CMAA) Specification #70, “Specifications for Top Running Bridge & Gantry Type Multiple Girder Electric Overhead Traveling Cranes,” 1975.
- Verification that the rating of components of the crane subject to degradation due to wear and exposure is approximately 15% higher than the load imposed by the reactor vessel head lift.
- Verification that non-redundant gearing meets the design standards of the American Gear Manufacturers Association (AGMA) as referenced in CMAA 70-1975, including the Crane Service Factors therein.
- Verification that the factor of safety for a non-redundant wire rope exceeds 10:1 for the reactor vessel head lift or the factor of safety exceeds 5:1 for the reactor vessel head lift and additional administrative controls confirm that the wire rope is in good condition prior to the lift.
- Administrative controls, including an additional crane operator positioned with effective access to the emergency stop button, to reduce the potential for operator errors and load hang-up events to cause a load drop.

The staff also found the specified equivalence measures acceptable. Operating experience demonstrates that special lifting devices are extremely reliable in holding the load once the lift has commenced (i.e., lifting device problems tend to be self-revealing at an early stage in the lift sequence). Credible failures in the handling system above the special lifting device include operator errors and control system failures that could lead to “two-blocking.” “Two-blocking” is a condition where the load block is brought into contact with the upper block, which often results in failure of the wire rope due to overstress in tension or shear. Other control system failures could result in an over-speed condition. Finally, wear and damage to moving parts have caused crane failures.

The staff finds the guidance contained in Section 3 of NEI 08-05 acceptable for classification of existing cranes used for reactor vessel head lifts as equivalent to single failure proof (i.e., a load drop need not be postulated due to the low frequency of handling system failure). The staff concluded that equivalence is appropriate for cranes used for this specific application because the reactor vessel is a robust structure and, thus, not vulnerable to drops from low heights. The proposed minimum hardware requirements (i.e., redundant hoist upper limit switches, a hoist over-speed sensor circuit, redundant holding brakes, and control stations that include spring-return-to-off switches and an emergency stop button) reduce the potential for control system failures or operator errors to result in a load drop. Enhancements to administrative controls governing crane maintenance, inspection, testing and operation further reduce the potential for failures that could result in load drops. In some cases, such as wire rope strength, enhanced administrative controls are used as an equivalence measure when physical changes to the crane are unreasonably expensive relative to the expected benefit. Therefore, the staff concludes that the proposed guidance for single failure proof crane equivalence for reactor vessel head lifts is acceptable.

3.4 Final Safety Analysis Report (FSAR) Updates

The industry initiative on control of heavy loads calls for licensees to provide a summary description of the basis for conducting safe heavy load movements in the next safety analysis report update. Section 4 of NEI 08-05, "FSAR Update," provides industry guidance for the content of FSAR updates describing the basis for safe heavy load movements.

Section 4 of NEI 08-05 provides a list of considerations for development of the FSAR update and an outline showing recommended format and content. Also, the guidelines recommend consulting NEI 98-03 guidance and ensuring the requirements of 10 CFR 50.71(e) are satisfied by the proposed update. The NRC staff reviewed the considerations and the proposed format and content. The staff found the guidelines complete, consistent with regulatory guidance related to the requirements of 10 CFR 50.71(e), and suitable for describing the basis for safe heavy load movements.

4.0 CONCLUSION

Based on the above review, the NRC staff has concluded that the guidelines contained in NEI 08-05, Revision 0, are acceptable for implementation of the industry initiative on control of heavy loads. With the exception that the NRC staff considers the acceptance criteria of Appendix F to the ASME B&PV Code, Section III, Division 1, (rather than the industry proposed strain-based criteria) appropriate for evaluation of coolant retaining component performance following a postulated head drop, licensees may consider the guidelines of NEI 08-05 as providing methods approved by the NRC for the specified applications when implementing the requirements of 10 CFR 50.59.