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INDUSTRY INITIATIVE ON CONTROL OF HEAVY LOADS

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Nuclear Energy Institute

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EXECUTIVE SUMMARY

On September 14, 2007, the industry's Nuclear Strategic Issues Advisory Committee approved an industry initiative to address NRC staff concerns regarding the interpretation and implementation of regulatory guidance associated with heavy load lifts. While there had been no significant events associated with heavy load lifts, NRC and industry identified a lack of consistency in plant licensing bases that pertain to this issue. The formal industry initiative specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices. An industry task force on heavy loads developed these guidelines to implement the initiative.

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BACKGROUND

On September 14, 2007, the industry's Nuclear Strategic Issues Advisory Committee approved an industry initiative to address NRC staff concerns regarding the interpretation and implementation of regulatory guidance associated with heavy load lifts. While there had been no significant events associated with heavy load lifts, NRC and industry identified a lack of consistency in plant licensing bases that pertain to this issue. The formal industry initiative specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices.

THE INITIATIVE

A. <u>For plants with an outage beginning before July 1, 2008:</u>

1) For all heavy load lifts, ensure 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 are adequately implemented and reflected in plant procedures.

2) For reactor vessel head lifts:

   a) If you have a single failure proof crane or 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, ensure your procedures for moving the head reflect your safety basis. Load drop analyses can be based on realistic (i.e. best estimate) calculations.

   b) If you do <u>not</u> have a single failure proof crane or a load drop analysis (generic or plant-specific) that bounds your planned lifts with respect to load weight, load height, and medium present, the head lift should be conducted "wet" (i.e., the maximum head lift height while over the refueling cavity should be the minimum necessary to clear immovable structures around the refueling cavity and the bottom of the head should be less than 15 feet above the refueling cavity water surface except where additional height is necessary to clear immovable structures once the cavity is fully flooded).

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

B. <u>For all plants with an outage beginning after July 1, 2008 and thereafter:</u>

1) For all heavy load lifts, ensure 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 are adequately implemented and reflected in plant procedures.

2) For reactor vessel head lifts and spent fuel cask lifts over the spent fuel pool, ensure you have a single failure proof crane or 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, and ensure your procedures for moving these loads reflect your safety basis. Load drop analyses can be based on realistic (i.e. best estimate) calculations.

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

4) In your next FSAR update, provide a summary description of your 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 FSAR.

5) If load drop analyses are used, ensure restrictions on load height, load weight, and medium present under the load are reflected in plant procedures.

IMPLEMENTATION

An industry task force on heavy loads was established to develop guidelines for the initiative. Section 1 of this document provides guidance on maintenance rule (a)(4) administrative controls. Section 2 provides guidance on reactor vessel load drop analysis, and Section 3 provides guidance on reactor head lift single failure proof crane equivalence. Several public meetings were held with the NRC to discuss aspects of Sections 2 and 3. In letters dated May 16 and May 27, 2008, the NRC stated its position on the industry guidance. These letters are included in the Appendix. Section 4 provides guidance on updating the FSAR.

1 MAINTENANCE RULE 10 CFR 50.65(a)(4) CONSIDERATIONS

1.1 INTRODUCTION

The heavy loads initiative requires utilities to ensure that their maintenance rule (a)(4) administrative controls include the movement of heavy loads as a configuration management activity. The maintenance rule, 10 CFR 50.65(a)(4), states:

"Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to those structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety."

A task force subgroup of industry experts developed the following guidance.

1.2 GUIDANCE

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). NRC Regulatory Guide 1.182 addresses use of this guidance.

Plants should address 10 CFR 50.65(a)(4) for heavy load lifts conducted during power operation, or during shutdown conditions. This would apply for reactor vessel head lifts, spent fuel cask lifts, and other heavy load lifts as defined under the plant's heavy load procedures. Most plants define heavy loads as loads greater than approximately 1000 pounds.

A quantitative risk assessment of the heavy load lift need not be performed. In general, quantifying the potential of a heavy load drop and its consequences is beyond the capability of quantitative risk assessment tools.

1.3 ASSESSMENT CONSIDERATIONS

The primary guidance involves communication between heavy lift activities and the work control (configuration management) process. These considerations include:

1. If a train, or equipment under the load path is protected (e.g., it is performing the safety function and its redundant counterpart is out of service for maintenance, and the risk management action includes protecting this function). Support equipment for the protected equipment should also be considered.

2. Whether the crane, hook, and rigging is single failure proof. (See Section 3 for determining single failure proof equivalence for the crane lifting the reactor vessel head.)

3. Whether the safety function is impacted by a potential load drop (in general this would be assumed to occur).

1.4 RISK MANAGEMENT ACTIONS

Risk management actions should be developed and implemented on the basis of the above considerations. These actions could include:

1. Revising the load path to preclude movement over the operating train, or conducting the heavy load lift at a different time, e.g., after redundant equipment has been restored to service.

2. Providing additional compensatory actions or backup safety functions to enhance redundancy of safety function performance during the heavy load lift.

3. Providing additional communication and awareness to operations and maintenance personnel of the load lift and its relation to maintenance activities.

4. Obtaining approval of plant management of the heavy load lift.

2 INDUSTRY CRITERIA FOR REACTOR VESSEL LOAD DROP AND CONSEQUENCE ANALYSIS

2.1 INTRODUCTION

A subgroup of industry specialists in load drop evaluations was formed to develop this section of the guidelines. The group was made up of personnel from licensees, Architect/Engineering firms, NSSS vendors and other specialty firms that have supported the industry from the time the initial NUREG-0612 guidelines were issued. This subgroup was tasked with developing criteria to perform realistic (i.e., best estimate) calculations and provide a document with sufficient detail to be useful both to the industry and the NRC. Care was taken to incorporate both lessons learned from previous analyses, as well as interface with the USNRC staff to provide criteria that allow a safe and realistic but also efficient and practical analytical process that will produce reasonable and acceptable results.

The purpose of these guidelines on reactor vessel head drop analyses is to demonstrate that after a postulated reactor vessel head drop accident, the core remains covered with coolant and sufficient cooling is available. It is not the intent of the industry initiative to endorse a specific methodology. However, it is important that general requirements for the analysis, material, modeling and acceptance criteria be available to provide consistency in plant licensing bases and for regulatory oversight.

This section provides a comparison of NUREG-0612 guidelines for analyses of postulated reactor vessel head drops and the industry initiative guidelines. This comparison makes clear the differences between the NUREG-0612 guidance and what is expected of realistic load drop analyses to conform to the initiative.

The NUREG-0612 comparison is followed by the detailed analysis guidance. As discussed above the initiative does not endorse a specific methodology. It does provide for consistency by including general requirements, material requirements, modeling requirements, and acceptance criteria. In addition, there is a discussion of parametric evaluations, in which a plant can be compared to another plant which has already performed an acceptable load drop analysis. Section 2.4 provides a technical basis for the guidelines.

2.2 COMPARISON OF INDUSTRY INITIATIVE WITH NUREG-0612 GUIDELINES FOR ANALYSES OF POSTULATED REACTOR VESSEL HEAD DROPS

Table 1 provides a comparison between the guidelines included in Section 5 and Appendix A of NUREG-0612 and the realistic analysis to be conducted for the initiative as they apply to reactor vessel load drop evaluations. In certain cases the initiative approach limits the scope of evaluation to cases that, based on previous evaluations, have been determined to represent worst case conditions. The criteria as clarified for the initiative analyses are satisfactory for future reactor vessel load drop evaluations.

TABLE 1
Comparison of Industry Initiative with NUREG-0612

5.1 Evaluation Criteria	Initiative Analysis
I. Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits);	Demonstrate that after the reactor vessel head drop, the core remains covered with coolant and sufficient cooling is available.
II. Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that keff is larger than 0.95;	
III. Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated); and	
IV. Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.	
Appendix A 1. General Considerations	**Initiative Analysis**
(1)That the load is dropped in an orientation that causes the most severe consequences	The reactor vessel head drop is concentric and impacts directly on the vessel flange.
(2) That fuel impacted is 100 hours subcritical (or whatever the minimum that is allowed in facility technical specifications prior to fuel handling)	N/A
(3) That the load may be dropped at any location in the crane travel area where movement is not restricted by mechanical stops or electrical interlocks	The reactor vessel head is dropped directly above the vessel at the maximum height controlled by plant procedures. In some plant procedures, the reactor vessel head may be moved horizontally and still be over the flange, and then lifted further. The maximum drop height is determined by the maximum height above the

	flange while the reactor vessel head center of gravity is still within the flange radius or over the flange. This height is used in the calculation of a concentric flat drop.
(4) That credit may not be taken for spent fuel pool area charcoal filters; if hatches, wall, or roof sections are removed during the handling of the heavy load being analyzed, or whenever the building negative pressure rises above (-)1/8 inch (-3 m) water gauge	N/A
(5) Analyses that rely on results of Table 2.1-1 or Figures 2.1-1 or 2.1-2 for potential offsite doses or safe decay times should verify that the assumptions of Table 2.1-2 are conservative for the facility under review. X/Q values should be derived from analysis of on-site meteorological measurements based on 5% worst meteorological conditions	N/A
(6) Analyses should be based on an elastic-plastic curve that represents a true stress-strain relationship	If the analyses are based on an elastic-plastic curve, it must represent a true stress-strain relationship.
(7) The analysis should postulate the "maximum damage" that could result, i.e., the analysis should consider that all energy is absorbed by the structure and/or equipment that is impacted	The analysis will consider the “maximum damage” caused by the transfer of energy to the vessel and supports. Analysis that accounts for appropriate consideration of conservation of momentum is acceptable. It is also acceptable to consider damping.
(8) Loads need not be analyzed if their load paths and consequences are scoped by the analysis of some other load	N/A
(9) To overcome water leakage due to damage from a load drop, credit may be taken for borated water makeup of adequate concentration that is required to be available by the technical specifications	To overcome water leakage due to damage from a load drop, credit may be taken for makeup water for BWRs and borated water makeup for PWRs of adequate concentration that is required to be available by the technical specifications.

(10) Credit may not be taken for equipment to operate that may mitigate the effects of the load drop if the equipment is not required to be operable by the technical specifications when the load could be dropped	N/A
Appendix A 2. Rx Vessel Head Drop Analysis	**Initiative Analysis**
*These guidelines only consider the dropping of the RV head assembly during refueling and do not apply directly to dropping of the reactor internals such as the steam dryer (BWR), moisture-separator (BWR) or the upper core internals (PWR); however, similar assumptions and considerations would apply to analyses of dropping of reactor internals.	Only Reactor vessel head drop is considered.
(1) Impact loads should include the weight of the reactor vessel (RV) head assembly (including all appurtances), the crane load block, and other lifting apparatus (i.e., the strongback for a BWR).	The analysis should include the weight of the reactor vessel (RV) head assembly below the hook.
(2) All potential accident cases during the refueling operation-. Areas of consideration as a minimum should be:(a) Fall of the RV head from it's maximum height while still on the guide studs followed by impact with the RV flange;(b) Fall of the RV head from its maximum height considering possible objects of impact such as the guide studs, the RV flange, the steam dryer (BWR) or structures beneath the path of travel; and(c) Impact with the fueling cavity wall due to load swing with the subsequent drop of the RV head due to lifting device or wire rope failure.	Area of consideration: Fall of the reactor vessel head from its maximum height allowed by plant procedures directly (concentrically and flat) on the vessel flange. In some plant procedures, the reactor vessel head may be moved horizontally and still be over the flange, and then lifted further. The maximum drop height is determined by the maximum height above the flange while the reactor vessel head center of gravity is still within the flange radius or over the flange. This height is used in the calculation of a concentric flat drop.
(3) All cases which are to be considered should be analyzed in the actual medium present during the postulated accident, e.g., for a PWR prior to reassembly of the reactor, the fueling cavity is drained after the head engages the guide studs to allow for visual inspection of the reactor core control drive rods insertion into the head. During this phase it should be considered that the head will only fall through air, without any drag forces produced by a water environment.	The analysis will consider the actual medium controlled by plant procedures.

(4) In those Nuclear Steam Supply Systems where portions of the reactor internals extend above the RV flange, the internals should be analyzed for buckling and resultant adverse effects due to the impact loading of the RV head. It should be demonstrated that the energy absorption characteristics (causing buckling failure) of these internals should be such that resultant damage to the core assembly does not cause a condition beyond the acceptance criteria for this analysis.	N/A
5) Reactor vessel supports should be evaluated for the effects of the transmitted impact loads of the RV head. In the case of PWRs where the RV is supported at its nozzles, the effects of bending; shear and circumferential stresses on the nozzles should be examined. For BWRs the effects of these impact loads on the RV support skirt should be examined.	All components and structures in the load path for the reactor vessel head drop will be evaluated to assure deformation is limited, that the core remains covered and that cooling of the core is maintained.
(6) The RV head assembly should be considered rigid and not experience deformation during impact with other components or structures.	The RV head assembly should be considered rigid unless explicitly modeled. The deformation of components attached to the RV head may be realistically considered.
Appendix A.4 Criticality Considerations	**Initiative Analysis**
4.1 Spent Fuel Pool Neutronics Analysis 4.2 Reactor Core Neutronics Analysis	N/A

2.3 GUIDELINES FOR REACTOR HEAD DROP DETAILED ANALYSES

The purpose of reactor vessel head drop analyses is to demonstrate that after a postulated reactor vessel head drop accident, the core remains covered with coolant and sufficient cooling is available.

These guidelines provide general requirements for the analysis, material requirements, modeling requirements and acceptance criteria.

It is not the intent of the industry initiative to endorse a specific methodology. It is important that the analyst responsible for the evaluation select the methodology that best addresses the specific issues at hand, that it is consistent with the analytical tools available and reflects the situation being evaluated. However, several methodologies have been used successfully in past analyses that are worthy of mention. These include the following:

- Finite Element Analysis (either the vessel and support system or possibly the head is included as an integrated model)

- Classical Analysis (typically used prior to the availability of complex FEA. These techniques may use closed form solutions, or an assemblage of mathematical expressions to represent the behavior of single or multiple components of the structure)
- Hybrid Analysis (portions of the total structure are represented by a series of FEA and/or mathematical expressions that are then combined as a total model through an assemblage of masses and complex springs)
- Parametric Comparative Analysis (a head-vessel/support system compared to a previously analyzed similar configuration by comparison of the individual parameters)

In many cases, licensees have already conducted load drop analyses. If these analyses have previously been approved by the NRC (for example, in safety evaluations) no further analysis is necessary. If not approved by NRC, the licensee may compare its previous analysis to these guidelines to determine if more analysis is needed. (For example, some classical analysis did not consider the need to look at the support structure underneath the vessel nozzles.)

These guidelines are written generically, not prescriptively, to provide acceptable methodologies and acceptance criteria. A reactor vessel head drop accident is considered to be a one-time "beyond design basis" accident scenario. To the degree possible, conservatisms are removed in an attempt to obtain the most realistic prediction of the outcome. Significant permanent deformation, displacement and damage to vessel supports, reactor nozzles and the reactor loop piping are acceptable outcomes as long as the core cooling criteria are met.

The analyses for reactor vessel load drop are extremely difficult and complex. It is expected that such analyses be performed only by highly experienced analysts that have an advanced knowledge of the complexities of dynamic impact based on nonlinear behavior of material, and are aware of material behavior well beyond yield strain and the uncertainties accompanying such an analysis. Attempts were made to incorporate the best available guidelines, based on a body of knowledge collected by industry experts over years of experience. All of the criteria presented are based on general observation of material behavior and as such, are believed to be well within expected material behavior for typical materials used in reactor vessel components and supports. There has been little need for national codes addressing load drop as this is not a common event in industry. Efforts are underway to codify load drop for spent fuel casks but have yet to be developed that are appropriate for reference. Both AISC and ACI have addressed impact to a measured degree. This information has been included in the criteria for those components and is the basis of certain criteria for load drop onto reactor components. However, the criteria presented below, in some cases, go beyond current ASME code limits, and directly applicable test results from actual load drops are not available to otherwise validate analysis. For this reason, there is a need for analysts capable of recognizing any uncertainty in the results and to address appropriate evaluation and resolution in the analysis process. Thus the importance of requiring knowledgeable and experienced analysts to perform this work. The resultant level of margins in the calculations needs to be commensurate with the uncertainties associated with the analytical methods being used.

2.3.1 General Requirements for Analysis

1. Structural elements in the impact load path from the reactor vessel flange down to the foundation mat need to be identified and evaluated.

2. The maximum potential energy for the head drop must be considered. The maximum potential energy is derived from the height of the head before it is dropped and the mass of the head assembly. The maximum height is the limiting value allowed by plant procedures. The dropped mass must include the head and all attachments below the hook. The actual medium through which the head is dropped (air or water) shall be considered.

3. The fall of the reactor vessel head is defined as a drop from its maximum height allowed by plant procedures impacting directly (concentrically and flat) on the vessel flange. In some plant procedures, the reactor vessel head may be moved horizontally and still be over the flange, and then lifted further. The maximum drop height is determined by the maximum height above the flange while the reactor vessel head center of gravity is still within the flange radius or over the flange. This height is used in the calculation of a concentric flat drop.

4. The evaluation may consider post-buckling response, as applicable. The stability of the vessel and support, after the load drop, must be ensured for deadweight.

2.3.2 Material Requirements

1. The representation of material behavior in the analysis shall be by true stress-strain curves. As an alternative, an elastic-perfectly plastic stress-strain curve may be used.

2. Material properties may be obtained from one of the following criterion:

   2.1 Use minimum code or specification yield and ultimate strength values for the affected components.

   2.2 Use representative or actual test data yield and ultimate strength for the affected components.

   It is acceptable to use curves developed from test data, which have similar engineering strengths and elongation as the average of engineering strengths and elongation from the component code or specification. As an alternative, it is acceptable to use true stress-strain curves for similar materials that have been modified to match the code or specification minimum properties for yield stress, ultimate stress and minimum elongation. When using actual test data, care must be taken to assure that the resultant data accounts for uncertainties caused by variations in properties throughout the “heat” of the material. Where multiple test results are available, the minimum values for both stress and strain should be used.

2.3 Sources for Material Properties

- CMTR's – Material test reports contain: Yield Stress, Ultimate Stress, Ultimate Strain or Rupture Strain, % of Area reduction at Rupture
- ASME Sec. II – Minimum CMTR required values (when CMTR's aren't available), Yield and Ultimate Stress (where CMTR minimum values are not available)
- ASME 2007 Section VIII, Division 2, Annex 3.D – General shape of Stress-Strain curves (note: Ultimate Strain and Rupture Strain utilize test data or CMTR's)
- ASM International – Atlas of Stress-Strain Curves; 1987 – Stress-strain curves to find Ultimate Strain and curve shapes for similar materials
- Battelle Pipe Fracture Encyclopedia; U.S. Nuclear Regulatory Commission; 1997; Stress-strain curves to find Ultimate Strain and general curve shapes for similar materials
- Other – Documented test data as available for the applicable material or a similar material
- Mechanical Behavior of Materials; by William F. Hosford; 2005 – pages 44-45; Engineering stress-strain to True stress-strain.

3. The design value or the minimum test data for 28-day concrete strength can be used with a strength increase due to aging.

4. A Dynamic Increase Factor (DIF) due to dynamic strain rate effects can be applied to the static stress strain diagram with appropriate technical justification. In some materials dynamic strain rate effects may cause a reduction in strain and, if so, shall be addressed when using strain based acceptance criteria.

5. Reference Sources to Establish Ultimate Strain (e_u)

- ASM International – Atlas of Stress-Strain Curves; 1987 – Stress-strain curves to find Ultimate Strain and curve shapes for similar materials
- Battelle Pipe Fracture Encyclopedia; U.S. Nuclear Regulatory Commission; 1997; Stress-strain curves to find Ultimate Strain and general curve shapes for similar materials
- Other – Documented test data as available for the applicable material or a similar material

This information is taken from tests. It may be necessary to find similar materials using material composition, CMTR's, and/or Code allowables (Sy,Su). The minimum value of Ultimate Strain from similar materials should be used in the equation for allowable strain.

2.3.3 Modeling Requirements

1. Computer codes that are known to be used and accepted throughout the industry for non-linear dynamic analysis applications and have been validated either against other analyses or test data are to be used for modeling applications. Finite Element model meshes are to be constructed with necessary detail and shape factors, consistent with accepted good practice, to assure that stress/strain results are representative of behavior from the actual structure. Selection of element type, as well as rules and guidelines are to be consistent with accepted literature requirements. One such guideline is found in the References sited in Section 4.5. Selection of elements is to also be consistent with those typically used for this type of analysis and known, from other experience, to produce representative and correct results. Modeling detail is particularly important in areas of high stress/strain.

2. NUREG-0612 required that the reactor vessel head be considered “rigid”. While this may be appropriate for simplified analyses, if accurately modeled the deformation associated with the head and any explicitly modeled attachments may be accounted for by the laws of physics. In cases where the head is not modeled explicitly, the head must be modeled as a rigid mass.

3. Large deformation option of the finite element code is used to account for the large deformations and strain associated with the drop event. The use of the large displacement option ensures that post-buckling behavior, necking (area reduction) and other instabilities are considered.

4. NUREG-0612 prohibited calculation of energy loss due to conservation of momentum. In a detailed elastic-plastic analysis a coefficient of restitution is not appropriate because any energy loss is modeled explicitly by the analysis. In simpler discrete mass models, a coefficient of restitution may be used if justification of the values is provided by the analysis.

5. NUREG-0612 required that all the energy be absorbed by the impacted structure. It is necessary to continue the analysis until the worst damage has been sustained by the impacted structure. This may include demonstrating that any further increase in displacement has ceased, the maximum strain deformation has been reached, or all the energy of the drop has been dissipated.

6. The reactor vessel model may include the mass and/or stiffness of the vessel contents if justified, or explicitly modeled.

7. The effect of the reactor loop piping may also be included depending on the complexity of the analysis and modeling methodologies. Including the reactor loop piping is at the discretion of the analyst.

8. For elastic-plastic finite element analyses, large deformation option of the finite element code shall be used to account for the large deformations and strain associated

with the analysis. This shall include deformations associated with post-buckling, if applicable, or concrete crushing, if applicable.

9. If the plastic deformation and friction at the contact surface are not explicitly modeled, a contact damping of 5% (steel) and 7% (concrete) of critical damping can be used in the analysis for those elements that remain elastic. The target critical damping values are consistent with the damping for welded steel and concrete structures for SSE events. Higher damping values may be used with an appropriate technical justification. The foundation mass and radiation damping may also be included.

10. The analysis of the drop should be modeled until it is demonstrated that the increase in displacement has ceased, the maximum strain deformation has been reached, or the energy of the drop has been dissipated.

11. When applied, concrete stiffness can be calculated by hand based on guidance in ASCE Standard 4, with appropriate consideration of edge distance. Alternatively, concrete stiffness can be calculated using an appropriate finite element model.

2.3.4 Acceptance Criteria

There are two general approaches to determining the acceptability of components and structures. These are:

- Equivalent Force Evaluations – This is based on determining an equivalent force that the structures must withstand in order for the impact to be resisted.

- Strain Based Evaluations -- This is based on the impacting structures absorbing a fixed amount of energy associated with the drop event. Energy is absorbed by elastic and plastic straining.

It is acceptable to have more than one approach used to evaluate the drop event. For example, steel support structures may be evaluated by an energy dependent approach while the supporting concrete may be evaluated based on the highest force during impact. However, it is not acceptable to evaluate forces using energy dependent strain acceptance criteria.

These criteria are considered an acceptable approach to demonstrating the suitability of the drop. Analysts may use other criteria, or other suitable design codes may be used to provide acceptance criteria, if adequate justification is provided.

1. Acceptance Criteria for Equivalent Force Evaluations

These criteria are applicable for evaluating the structure for the maximum forces. The criteria are applicable to cases where the stability is being demonstrated by force balance. Any of the methods permitted by these standards for evaluation of forces are acceptable.

1.1 Coolant retaining components and supports may be shown to be acceptable using the acceptance criteria provided in the ASME B&PV Section III, Appendix F. Note that the DIF factor should not be applied to the allowable stress when using this acceptance criterion.

1.2 Concrete structures may be evaluated using the requirements of ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary", American Concrete Institute. A capacity reduction factor of 1.0 may be used for bearing. For concrete, the use of ductility-based methodology as provided in ACI-349-01 is also appropriate. Other Standards and references that may be considered include the military Tri-services Manual (TM5-1300) and ASCE Manual 58.

2. Strain Based Acceptance Criteria

These criteria are applicable for applications using energy methods. The maximum strain is determined based on the deformation resulting from defined impact energy. Alternately, it is also acceptable to use the criteria listed above. Strain criteria are only applicable to materials with $Sy/Su < 0.7$.

When using strain based acceptance criteria, the effects of biaxial and triaxial tensile strains shall be considered.

If a DIF is used (to increase yield and/or ultimate strength) the impact on allowable strain needs to be considered.

2.1 Strain Acceptance Criteria for Coolant Retaining Components

- Average (through thickness) equivalent total strain is limited to strain of $0.5\ \varepsilon_u$, where ε_u is the strain at ultimate stress (see Figure 1).
- Average plus linearized (through thickness) equivalent total strain is limited to strain of $0.75\ \varepsilon_u$.

It is anticipated that the application of this criteria will prevent, or at least control potential leakage to limits well within the ability of the container to maintain coolant over the core. However, attention is to be given to evaluate the highest strained areas to assure leakage will be prevented, or at least controlled. The criteria allow for linearized strains consisting of average through thickness plus bending to be increased by 50% over the average at the extreme outside fiber. This is intended to impose additional control on the maximum strain within an element, is expected to be very much localized and, well within the increases for bending over average through thickness strains allowed by AISC A690 for extreme loading cases. However, the analyst needs to evaluate the results of the analysis and his knowledge to assure that, regardless of the strain rate, leakage control can be maintained within acceptable limits (e.g. any potential for cracking will remain localized and that make-up from other sources can maintain the core cool and covered).

2.2 Strain Acceptance Criteria for Component Supports

General: The following criteria are used to determine the adequacy of individual load carrying members of a support system. Load carrying members which meet these criteria are shown to be fully effective during and after the load drop event.

Most support configurations consist of numerous redundant load carrying members. If permitted by the system's redundancy, individual members, or parts of a member, may exceed these criteria as long as the global effect of the exceedance is considered in the analysis and the transmitted energy of the drop is shown to be absorbed by redundant structures. Such cases shall be identified and justified in the analysis.

Steel Supports:

- Tensile strain – structural members loaded predominately in tension shall limit average total strain to 0.5 ε_u where ε_u is the strain at ultimate stress (see Figure 1).
- Tensile and bending –The strain of interest for bending members is on the flange (or outer edge member). The ratio of the thickness of the flange (or outside plate element) to the depth of the bending member is typically a relatively small number so that there is very little variation in strain through the thickness of the "flange". For this reason it is acceptable to combine the average tensile strain due to moment loading with other axial strains as average through thickness strains. The combined tensile plus bending strain shall be less than 0.75 ε_u. Also, the analysis must consider potential for lateral and torsional buckling.
- Compressive loads – members loaded in compression need to be evaluated for potential elastic and plastic instabilities. This may be demonstrated using numerical methods capable of determining large deformation behavior. Such methods need to conservatively consider structural features which could reduce stability.
- Pure shear – The average primary shear stress across a section loaded in pure shear shall not exceed $0.6S_u$ unless it is bolt material, in which case the criteria for bolts and pins will take precedence.

Welded Structures

- Full penetration structural welds shall be considered equivalent to the base material. Weld metal is typically of greater strength than the parent steel.

Because of this potentially increased strength, there may be a reduction in strain capacity. Weld areas where strains are higher than conventionally allowed are to be evaluated to assure that computed strain is within the strain limits for the weld material.

- Fillet and other partial penetration welds required to maintain the support integrity shall be treated on a case-by-case basis with appropriate reduction of stress or strain capacity that account for the lack of ductility of the weld. The AISC Manual of Steel Construction, 9th Edition, page 4-71 is one source that provides load resistance curves for certain fillet weld loading configurations that may be useful in determination of fillet weld capacity.

Bolts and pins

- High strength bolting material loaded in tension or bending shall meet the applicable stress limits of ASME Section III, Appendix F or, as applicable, AISC N690.

Bearing stress

- Bearing stress in steel structures need not be considered for this event.

Concrete Supports:

Bearing Under Plates

- Requirements of ACI 349-97 will be met with capacity reduction factor of 1.0. When bearing capacity under a highly stressed portion due to bending is exceeded, the analysis shall consider the effects of crushing of the highly stressed portion.

Overall Response of Walls and Piers

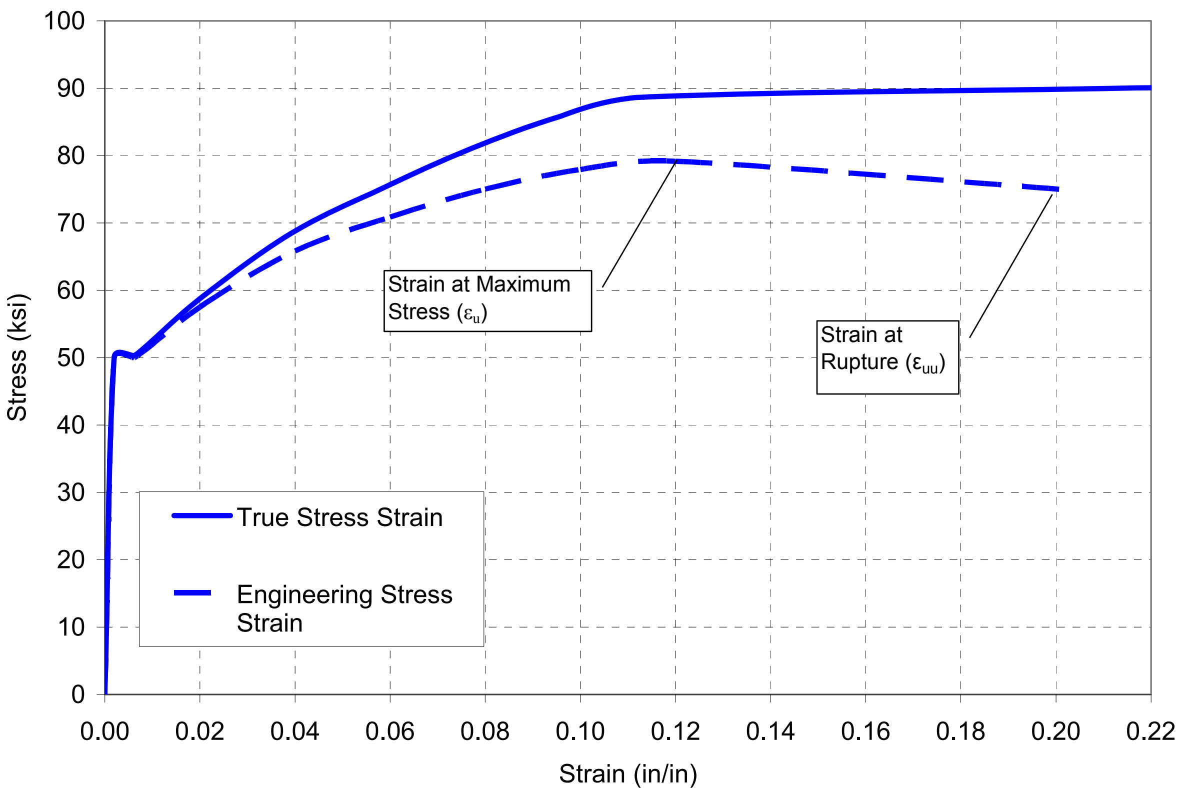
- The requirements of ACI 349-97 will be met with a capacity reduction factor of 0.9.

3. Maximum Deflection

3.1 The maximum vertical deflection of the reactor vessel under the initial reactor head impact shall be less than the acceptable deflection limit, which is necessary for the RCS attached piping to supply coolant to maintain the core flooded and prevent boil-off.

3.2 A support or concrete structure can fail one of the acceptance criteria, but the failure must be shown to be displacement limited, and the piping needed to maintain long-term cooling must be shown to remain leak tight for the imposed additional displacement.

Figure 1. Definitions of Strains



4. Parametric Evaluations of Reactor Vessel Head Drop and Consequence Analyses

Parametric evaluations have been successfully performed in the past and remain a valuable tool to evaluate the consequences of a load drop for a plant that has sufficiently similar reactor vessel configurations, including the support arrangement. Critical parameters to be included in a parametric analysis are listed below. While the objective is to demonstrate that the object of the evaluation has at least as much margin to assure the core remains covered and cooling is available after the postulated event, it is not necessary that each parameter of the evaluation envelope that of the source evaluation. It is the responsibility of the analyst performing the evaluation to assure the effect of the individual parameters is properly weighted. Parameters for consideration include:

- Drop Height
- Reactor Vessel Head Weight - mass of dropped objects (head plus equipment below the hook)

- Strength (yield and ultimate stress) and ductility limits of the vessel and support material and/or strength of concrete
- Geometry of the vessel affected by the drop
- Support configuration
- Dimensions (size, distance from the reactor) of the supporting components and structure

5. References

To further provide guidance for future reactor vessel head drop analyses, a list of references commonly used, at least in part, are identified. Care must be taken in the use of these references to assure they are limited to specific portions of evaluations that are specifically related to load drop analysis. For example, ASME Section III, Division 1, Appendix F provides excellent criteria for Level D Service Limits, which is appropriate for load drop considerations. However, the criteria in this Appendix are based on time-dependent external or internal body force loads and not impact loading. Therefore the analyst needs to be selective in the application of this code to assure that references to load limits do not improperly limit available capacity to resist impact loads.

- ASME Section III, Division 1, Non-mandatory Appendix F
- ANSI/ANS-58.2, 1988, Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture
- ANSI/AISC N690-1994 [Q1.5.8] including supplement 2 (2004)
- ACI-349-1997, Appendix C (supplemented via RG 1.142 Regulatory Positions 10 and 11 as they apply to impact loading)
- Finite Element Mesh Considerations for Reduced Integration Elements, Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials, PATRAM 2007, Bjorkman, Gordan and Piotter, Jason

2.4 TECHNICAL BASIS

2.4.1 General Requirements for Analysis

The analysis of the system needs to include all appropriate elements in the impact load path from the reactor head down to the foundation. There are a wide variety of configurations so the specifics are left to the analysts.

The head drop must be based on the maximum head drop height and include the head and all attachments. The guidance in NUREG-0612 originally required that the weight of the hook and the load block assembly be included. Subsequent detailed evaluations have determined that in a realistic event, there is a time-delay between the drop of the load and the hook and it is appropriate to not include the hook and the load block.

After the load drop event components and structures may be severely damaged. Therefore, the stability of the vessel and support, after the load drop, must be ensured for deadweight.

2.4.2 Material Requirements

True stress strain curves are required by NUREG-0612. True strain is necessary to accurately account for the large strains during a large deflection analysis.

Code or specification material properties may be used to provide physical properties for the specific heat of material and product form used to fabricate a component. The properties are typically obtained using ASTM mechanical test procedures for the specific material type and grade including the number and location of tensile test specimens. These procedures were developed to obtain acceptable representative mechanical properties which can be used to certify that the material meets the minimum specifications. Therefore fabricators tend to produce material which has nominal specifications which exceed the minimum based on their experience. For ASME Code design work, the Code minimum properties must be used to show Code compliance for specified design and service condition loadings. However, for evaluation of conditions which occur during operating service, the Code recognizes that other mechanical properties are more appropriate to apply.

The design value or the minimum test data for 28-day concrete strength can be used with a strength increase due to aging. The increase in concrete strength beyond 28 days is a well established characteristic of concrete.

A Dynamic Increase Factor (DIF) due to impact loading can be applied to the stress ordinate of the static stress strain diagram if based on an appropriate technical justification.

2.4.3 Modeling Requirements

See Section 3.0

2.4.4 Acceptance Criteria

The acceptance criteria have been divided into equivalent force evaluations and energy balance evaluations. This is the same approach used in the design of pipe whip restraints discussed in ANSI/ANS 58.2. This standard recognizes that there are different failure mechanisms associated with each type of analysis and a different acceptance criterion is required for each. As noted, detailed analyses can be a mix of the two approaches.

The evaluation of forces is based on equilibrium by force balance. These forces loading a structure are not limited by displacement or energy and failure occurs when the stress exceeds ultimate, regardless of the strain.

Energy balance evaluations are evaluations that are based on a structure deforming and absorbing a fixed amount of energy associated with the event. These are also referred to as energy limited evaluations. A head drop is an energy limited event because only a fixed amount of energy is available to deform the impacted structures. This fixed energy is absorbed by elastic and plastic straining. For tensile loadings, energy will be absorbed by plastic deformation until the material reaches rupture strain, not ultimate stress. For compressive loadings, ductile material will continue to maintain a load capacity until some form of instability such as global or local inability to support the weight of the vessel and head plus the head package occurs.

Acceptance Criteria for Equivalent Force Evaluations

Current ASME Section III Appendix F (Code) acceptance criteria are based on the concept of limiting the loads and/or stress to a percentage of the minimum ultimate tensile strength. This provides adequate margin for a force based conditions. If the force/stress exceeded the limit deformation would continue until catastrophic failure occurs. These criteria were originally developed for conditions in which the load is not limited, like a pressure or deadweight load.

For a typical load based event, margin to failure can be understood in terms of load or stress level. The Code margin is approximately 30% to failure for primary membrane stress (.7Su) and 10% to local failure due to maximum primary stresses (.9Su).

Strain Based Acceptance Criteria

These criteria are applicable to cases where the stability is being demonstrated by an energy balance. The maximum strain is determined based on the deformation resulting from a defined and limited impact energy. The margin to failure is based on the difference between the permitted strain limit and when the material reaches strain at ultimate stress.

For coolant retention components, the strain limits provide greater margins (in terms of strain margins) to ASME Code Section III Division 1, Appendix F (30% and 10% as discussed above). This indicates that the strain criteria are not excessive or beyond accepted Code margins on general or local failures.

For supports the strain limits provide significant margins on either general or local failure. Large-deformation analysis and true stress-strain is used so that calculated strains are realistically calculated. Also, nuclear grade material brittle fracture is not a concern. By setting the acceptance limit at ultimate strain considerable margin against initiation of cracking due to over straining is assured

In any case the maximum vertical deflection of the reactor vessel under the reactor head impact shall be less than the acceptable deflection limit that is necessary for the RCS attached piping to supply coolant to maintain the core flooded and prevent boil-off.

Strain energy methods have been utilized in energy balance type analyses for faulted accidents other than head drop. These accidents are similar to head drop in that they are faulted high velocity transient analyses. The standards used for these analyses are provided below for information only:

ANSI/ANS-58.2-1988 sections 6.5 and 6.6
AISC std. N690-1994 [Q1.5.8] including supplement 2 (2004)

3 REACTOR HEAD LIFT SINGLE FAILURE PROOF CRANE EQUIVALENCE

3.1 INTRODUCTION

The industry initiative on Control of Heavy Load Lifts provides the alternative of having a "single failure proof crane" in performing reactor vessel head lifts and spent fuel cask lifts over the spent fuel pool. This section provides industry guidance for determining single failure proof equivalence for cranes for the limited purpose of lifting the reactor vessel head. It 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 construction plants.

3.2 PURPOSE AND BACKGROUND

Generic Letter 80-113 (supplemented later by Generic Letter 81-07) provided the methodology for addressing Single Failure Proof Handling Systems; see Attachment 1 of enclosure 3 of the generic letter. Item 2 of Attachment 1 asked for a detailed point-by-point comparison of the crane in question to NUREG 0554, Single Failure Proof Cranes for Nuclear Power Plants. Some utilities submitted Phase II responses that included this comparison. For any gaps found in those NUREG 0554 comparisons, the equivalence measures provided by this guidance document can be used to fill those gaps and allow for a reactor head lift in accordance with the Heavy Loads Initiative.

For those utilities that did not pursue the Single-Failure-Proof option with their Phase II responses or were licensed after Generic Letter 85-11, the equivalence measures provided by this guidance document may be applied to a reactor head lift if the crane used to make the lift is equipped with certain safety features and also has key supporting documentation. The minimum safety features are as follows:

- 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

The key supporting documentation includes the following:

- Calculation to show the crane is capable of holding the load during a Safe Shutdown Earthquake or an Event Frequency Calculation to show that the frequency, based on return period of SSE and time the load is over the reactor vessel, is <1E-7 per year
- Calculation to show the Crane meets the CMAA #70-1975 allowable stresses for the bridge, end trucks, and trolley structural components

- Calculation which shows the rating of components of the crane subject to degradation due to wear and exposure is approximately 15% higher than the Maximum Critical Load (in this case, the head lift) lifted by the crane
- For cranes with a single-hoist drive unit (ASME NOG-1, Figure 5416.1-1), a calculation that documents the gearing meets the design standards of the American Gear Manufacturers Association (AGMA) as referenced in CMAA #70-1975, including the Crane Service Factors therein
- Calculation that determines the Factor of Safety on the Wire Rope for the head lift. Factor is determined by multiplying catalog breaking strength of rope by the number of parts of line and dividing by MCL rating plus weight of block:

$$\mathrm{FS} = \frac{\text{Rope Breaking Strength} \times \text{Parts of Line}}{\text{MCL Rating} + \text{Weight of Load Block}}$$

With these minimum features, the supporting documentation, and the inclusion of the additional control measures described in this guidance document, the lifting of the reactor head can be made in accordance with the Heavy Loads Initiative.

The lifting devices used below the hook to make the reactor head lift are required to meet the Phase I requirements as delineated in NUREG 0612, Section 5.1.1.(4).

3.3 METHODOLOGY

The NEI Single Failure Proof Crane Equivalence subgroup reviewed Appendix C of NUREG-0612, Modification of Existing Cranes. This appendix references NUREG-0554 and summarizes the single failure proof guidelines in ten specific areas. The appendix states that, "In the case of a new crane, all the recommendations contained in NUREG-0554 should be followed; however, in the case of an existing crane that is to be upgraded to the guidelines of Section 5.1.6, space economies for the crane may not allow ready application of all the safety features to the crane. Additionally application of certain other features may not be practical since they would require replacement of certain components whose adequacy can be verified by alternative measures. Thus, certain adjustments may be necessary to compensate for those features that will not be included." The appendix then provides some examples of alternative approaches.

The subgroup reviewed section 5.1.6, Single-Failure-Proof Handling Systems of NUREG-0612 which states that the purpose of a crane upgrade is "to improve the reliability of the handling system through increased factors of safety and through redundancy or duality in certain active components."

The subgroup also reviewed Generic Letter 85-11, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612, dated June 28, 1985. The team noted the NRC determined that upgrading cranes to single failure proof was not cost beneficial, and that the Phase I activities had significantly reduced the risk of drops of heavy loads such that Phase II actions were not required.

Given the regulatory bases described above, the subgroup evaluated the ten specific areas listed in Appendix C of NUREG-0612 to determine what reasonable measures could be taken by plants to achieve an equivalent single failure proof crane.

Table 1 summarizes the ten key crane design and hardware equivalency guidelines. The "Bases" column designates whether equivalent measures are permitted under the industry initiative. An "E" indicates the equivalency measure is acceptable; an "MR" indicates the crane must meet the Design/Hardware requirement. The hardware requirements of Table 1 are the minimum safety features needed to achieve single-failure-proof equivalence.

Table 1 Crane Equivalency Guidelines - Design/Hardware Requirements

Equivalency Guideline	Design/Hardware Requirement	Bases
(9) Operator Error	Master Switches with Spring Return to Off Feature Cab Mounted Emergency Stop Button	MR MR
(6) Load Hang-Up/ Overspeed	Overload Sensor/Circuit Overspeed Sensor/Circuit	E MR
(7) Two-Block	Two Upper Limit Switches (Second Upper Limit shall be a Power Disconnect)	MR
(5) Wire Rope	Dual Wire Ropes	E
(4) Control Design	Master Switches with Spring Return to Off Feature Cab Mounted Emergency Stop Button Two Holding Brakes	MR MR MR
(1) Stress Limits	CMAA 70-1975 Stress Limits	MR
(3) Earthquake	Calculation for SSE with MCL	E
(2) Material	Toughness Properties Known and T_{min} Established	E
(10) Material	Cold-Proof Test	E
(8) Drum	Drum Safety Plates	E

3.4 EQUIVALENCE

Table 1 indicates whether equivalent measures are available for the crane lifting the reactor vessel head. Implementation of the equivalent measures is described in Table 2, which lists the Equivalency Guideline, the key safety issue being addressed, and the equivalent measures required to satisfy the industry initiative. Table 2 lists the safety issues in order of significance, where significance means the most likely cause of crane failure, and the area of greatest gain in reducing the risk of crane failure and load drop. The three most significant safety issues are operator errors, load hang-up, and two-blocking. These "operational" safety issues are also the likely cause of structural challenges to the crane. The remaining seven safety issues require additional safety measures or higher factors of safety which are designed to mitigate the destructive effects of an operational safety event. These seven safety issues are areas of lower probability of failure and are not listed in any order of significance.

The subgroup believes that the objective of the single failure proof crane equivalence can be achieved by reasonable, cost-effective preventive measures for the first three safety issues. These are the first barrier to failure and can be considered preventive measures because if successful, they prevent the high levels of stress that the next seven safety issues are designed to mitigate.

Table 2 Crane Equivalency Matrix

Equivalency Guideline	Safety Issue	Equivalence Measures	Significance
(9) Safety devices such as limit switches provided to reduce the likelihood of a malfunction should be in addition to those normally provided for control of maloperation or operator error.	Prevention of Operator Based Errors	**Crane used to lift Reactor Head:** • Shall be equipped with Master Switches with Spring Return to Off Feature and • Shall be equipped with a Cab Mounted Emergency Stop Button (ESB) within reach of a Crane Operator. This ESB must be separate or unique from a radio or cab operated control switch and • Performance of a Pre-Operational Check[1] **Lifting of Reactor Head shall have the following administrative controls in place:** • 3-Way Direct Communications established between Crane Operator, Person-in-Charge, and Signal Person via headsets and • Second Crane Operator placed in cab of crane, or the most effective location with access to an ESB, to act as an observer/ spotter unless the crane is equipped with floor mounted ESBs that are manned during lift performance and • Backup Emergency Stop Signal such as an air horn (pre-tested) provided in case of loss of direct communication and • Pre-Job Brief performed that includes identification of Supervisory Oversight, Establishment of Lift Management Protocol, Acceptable Travel Limits of Crane, Verification of ESB Locations, and Manning of ESBs. and • Maintenance Rule (a)(4) Measures addressed in Outage Safety Plan	High - 1

Equivalency Guideline	Safety Issue	Equivalence Measures	Significance
(6) Sensing devices should be included in the hoisting system to detect such items as overspeed, overload, and overtravel and cause the hoisting action to stop when limits are exceeded.	Elimination of Load Hang-up / Overspeed Type Events	**Crane used to lift Reactor Head:** • Shall Be Equipped with an Overload Sensor/Circuit[2] **OR** • Have a Load Cell / Load Pin provided either on Crane or as part of Lift Rig and • An Individual designated to Observe Load Cell and Confirm Load is less than the weight allowed by plant procedures. Observations shall continue until lift has cleared all potential hang-up points **OR** • Spotters placed at critical locations to monitor lift and observe potential binding and • Spotters equipped with a means of transmitting an emergency stop signal and • Floor mounted ESB's that are manned during lift **Crane used to lift Reactor Head:** • Shall be Equipped with an Overspeed Sensor/Circuit[3] and • Shall have a Pre-Lift Check[1] prior to the lift	High - 2
(7) The reeving system should be designed against the destructive effects of two-blocking.	Elimination of a Two-Block Event	**Crane used to lift Reactor Head:** • Shall be equipped with Two Upper Limit Switches (Second Upper Limit shall be a Power Disconnect)[4] and • Upper Limit Switches must be checked during the station specific crane inspection program in accordance with the requirements of ANSI B30.2-1976. and • Second Crane Operator placed in cab of crane, or the most effective location, to act as an observer/ spotter	High - 3

<table>
<tr><th>Equivalency Guideline</th><th>Safety Issue</th><th>Equivalence Measures</th><th>Significance</th></tr>
<tr><td>(5) Design of the wire rope reeving system should include dual wire ropes.

Note: Dual wire ropes means the hoist is equipped with redundant reeved wire ropes.</td><td>Dual Wire Rope[5]</td><td>Crane used to lift Reactor Head:
• When equipped with a single wire rope, shall have a calculation that shows the wire rope factor of safety for the MCL is 10:1 or greater
OR
• When equipped with dual wire ropes, shall have a calculation that shows the wire rope factor of safety for the MCL is 5:1 or greater on each of the wire ropes
OR
• When equipped with a single wire rope, shall have a calculation that shows the wire rope factor of safety for the MCL is greater than 5:1 and less than 10:1
and
• Shall have the wire rope inspected prior to the lift with the maximum allowance for broken wires being -
o For running ropes, six randomly distributed broken wires in one lay or two broken wires in one strand in one lay.
o For rotation-resistant ropes, two randomly distributed broken wires in twelve rope diameters or four randomly distributed broken wires in sixty rope diameters
and
• Shall have the wire rope inspected prior to the lift with the maximum allowance for wear being -
o One-Sixth of the original diameter of the outside individual wires[6]
o For ropes over 3/4” and through 1-1/8” diameter, a maximum allowable reduction in overall rope diameter of 1/32”
o For ropes over 1-1/8” and through 1-1/2” diameter, a maximum allowable reduction in overall rope diameter of 3/64”
and
Perform a 5 minute hold of the load after the initial lift is made. (Full expected weight of the head. It is important to get the full weight. Once the lift is made, it may take several more inches to ensure the head is full up.)</td><td>Low</td></tr>
</table>

Equivalency Guideline	Safety Issue	Equivalence Measures	Significance
(4) Automatic controls and limiting devices should be designed so that component or system malfunction will not prevent the crane from stopping and holding the load safely.	Control Design	**Crane used to lift Reactor Head:** • Shall be equipped with a Cab Mounted Emergency Stop Button within reach of the Crane Operator. This ESB must be separate or unique from a radio or cab operated control switch. and • Shall be equipped with Master Switches with Spring Return to Off Feature and • Shall be equipped with two Holding Brakes[7] and • If equipped with Electric Holding Brakes, ensure the design does not release the brake after restoration of power from the loss of power event	Low
(1) The allowable stress limits should be identified and be conservative enough to prevent permanent deformation of the individual structural members when exposed to maximum load lifts.	Stress Limits	**The crane used to lift Reactor Head shall have:** • A calculation to show the bridge, end trucks, and trolley structural components crane meet the allowable stresses of CMAA #70-1975 or alternative specification as stated in NUREG 612 Section 5.1.1(7) and • A calculation which shows the rating of components of the crane subject to degradation due to wear and exposure is approximately 15% higher than the maximum critical load lifted by the crane [8] and • For cranes with a single-hoist drive unit (ASME NOG-1, Figure 5416.1-1), a calculation that documents the gearing meets the design standards of the American Gear Manufacturers Association (AGMA) as referenced in CMAA #70-1975, including the Crane Service Factors therein	Low

Equivalency Guideline	Safety Issue	Equivalence Measures	Significance
(3) The crane should be capable of stopping and holding the load during a seismic event equal to a Safe Shutdown Earthquake (SSE) applicable to that facility.	Earthquake	**Crane used to lift Reactor Head:** • Shall have a calculation to show the crane is capable of holding the load during an SSE **OR** • Shall have an Event Frequency Calculation to show that the frequency, based on return period of SSE and time the load is over the reactor vessel, is <1E-7 per year.[9]	Low
(2) The minimum operating temperature of the crane should be determined from the toughness properties of the structural material and that are stressed by the lifting of the load.	T_{min} for Operation	**Crane used to lift Reactor Head:** • Shall have a cold-proof load test **OR** • Shall be operated in an environment where the ambient temperature in the vicinity of the crane is at least 70° F or greater (Verify prior to lift)[10]	Low
(10) The crane system should be given a cold proof test if material toughness properties are not known	T_{min} for Operation	**Crane used to lift Reactor Head:** • Shall have a cold-proof load test **Or** • Shall be operated in an environment where the ambient temperature in the vicinity of the crane is at least 70° F or greater (Verify prior to lift)[10]	Low
(8) The hoisting drum(s) should be protected against dropping should its shafts or bearings fail.	Drum Safety Plates[5]	**Crane used to lift Reactor Head:** • Ensure all credible failure modes have been eliminated by the use of measures outlined above[11] and • Inspect drum bearings before lift as part of the station equivalent of an ASME B30.2-1976 Periodic Inspection of the Crane	Low

[1] Crane Inspections are defined as follows:

- Pre-Operational Check - Station specific inspection program in accordance with the requirements of ANSI B30.2-1976 performed at the start of the refueling outage. This inspection must include a detailed inspection of the wire rope (for cranes with a single wire rope and FS between 5:1 and 10:1, see Equivalency Guidance Item 5 for inspection criteria) and drive train. All safety functions included on the crane (limit switches, overspeed, overload, etc) must be verified as functional during this inspection.
- Pre-Lift Check - Prior to the lifting of the reactor head, another functional check of the crane safety features and braking systems shall be performed.

[2] Overload Circuit - If the hoist used to lift the reactor head is equipped with an overload circuit, the overload circuit must meet the following requirements:

- Switch shall trip when the load on the hook exceeds 125% of the design rated load.
- Operation of the overload trip switch shall remove power from the hoisting motor and cause all holding brakes to set.
- Require manual reset.

[3] Overspeed Circuit - The hoist used to lift the reactor head must have an overspeed circuit which meets the following requirements:

- Switch shall trip when the hook lowering speed exceeds the vendor recommended percentage (typically 115% to 125%) of the design rated load lowering speed.
- Actuation of the overspeed trip switch shall cause all holding brakes to set.
- Require manual reset.

[4] Second Upper Limit Switch - This needs to be done via contactor for Variable Frequency Drive (VFD) equipped cranes to protect the drive against voltage spikes.

[5] Significant modifications may be necessary in order to add redundant reeving or drum restraints to an existing hoist

[6] For hoists that have been equipped with compacted wire ropes, the wear shall be one-sixth of the original diameter of the outside individual wires or ½ the manufacturers recommended wear allowance; whichever is less

[7] Holding Brake - The hoist used to lift the reactor head must have a holding brake system which meets the following requirements:

- Two independent holding brakes are required, with each brake rated for at least 125% of the head lift hoisting torque at the point of brake application.
- For cranes equipped with a single-hoist drive unit, the holding brakes shall be mounted on opposite sides of the gearbox or so arranged that a single coupling failure does not de-couple both brakes from the gearbox.
- For cranes equipped with dual (redundant) hoist drive units (single drum), each gearbox shall be equipped with a holding brake.
- The holding brakes on hoists shall be applied automatically when power is removed from the hoist
- A brake which acts directly on the wire rope drum or its shaft is considered a holding brake.

[8] As an alternative to 15% margin, for cranes with a margin greater than 8% and less than 15%, the design of the crane must be equivalent to CMAA Class C (Moderate Service) or higher (Polar Crane usage is equivalent to CMAA Class A1 or Standby Service). For those cranes with margins of 8% or less, the design of the crane must be equivalent to CMAA Class D or higher.

[9] Event Frequency Calculation: The SSE return period for plants ranges from about 10^5 to about 10^4 years.
For the purpose of this calculation use a plant specific SSE return period. The event frequency calculation would be:

(1/site specific return period) X (hours reactor vessel head is above the vessel (lifting and replacing)/ approximate hours between refuelings)

Examples for a plant with an SSE return period of 10^4 years:

Hrs Head is above vessel	Event Frequency	
Months between cycles	18	24
Two	1.52E-8	1.14E-8
Four	3.04E-8	2.28E-8

Note: This is an approximate calculation. It is NOT required to measure actual times the head is above the vessel during actual head movement. The approximate time above the vessel is based on experience and procedures. If procedures or experience change such that the time above the vessel increases significantly, a recalculation would be appropriate. This approach is consistent with ASME NOG-1-2004.

[10] Minimum Ambient Temperature - In lieu of a 70° F minimum temperature, test coupons may be taken from the primary load-bearing crane structural members (e.g. bridge girders, trolley structure, upper and lower block frames) to allow for impact testing as described in NUREG 0554, Section 2.4. The results of this testing may allow the establishment of a lower minimum operating temperature. (The Quad Cities Station successfully lowered their minimum operating temperature through such testing.)

[11] Failure modes – The credible failure mode is overload which includes load hang-up and two-block. The load hang-up and two-block events are addressed by other design features or equivalence measures.

3.5 EQUIVALENCE EXAMPLES

Table 3 and Table 4 provide examples of how the equivalence measures can be used to show that a reactor lift meets the intent of the Heavy Loads Initiative.

Crane Originally Evaluated for NUREG 0612 Phase II

The Table 3 example is for a crane that was evaluated as part of the Phase II response to NUREG 0612. This evaluation consisted of a point-by-point comparison to NUREG 0554. The results of the point-by-point comparison indicated the crane did not meet NUREG 0554 in two areas. These were:

1. No fracture toughness properties known or a cold proof test performed
2. Dual wire-rope with a FS less than 10:1

Attachment C of NUREG 0612 provided the utility with an alternative to performing the cold proof test. However, a hardware modification was required to address item 2. With the issuance of GL 85-11, any planned hardware modifications to correct item 2 were deemed as no longer required.

As a result of the Heavy Loads Initiative, the utility has taken another look at the original Phase II point-by-point comparison. This guidance document provides equivalency measures that address both of these areas. Table 3 shows how the point-by-point comparison was revised to take credit for these equivalency measures. After updating the point-by-point comparison using the equivalency measures, the comparison was documented in accordance with station procedures.

Crane Never Evaluated for NUREG 0612 Phase II

The Table 4 example represents a crane that was NOT evaluated as part of the Phase II response to NUREG 0612. Thus, there is not point-by-point comparison to NUREG 0554.

As a result of the Heavy Loads Initiative, the utility has reviewed the crane design and existing documentation. Based upon this review and the equivalency measures provided by this guidance document, the utility has determined that pursuing a single-failure-proof equivalency for this crane is the best approach for the reactor head lift. To show single-failure-proof equivalency, the utility developed a step-by-step plan which describes the crane action being performed, the safety issue(s) that apply to the crane action, and the checks and controls taken to address the safety issue(s). These checks and controls are based upon the equivalence measures provided by Table 2.

In the Table 4 example, the crane used to make the lift has the following design features:

- No fracture toughness properties known or a cold proof test performed
- A single wire-rope with a FS between 5 and 10
- Master Switches with Spring Return to Off Feature

- Two Upper Limit Switches (Second Upper Limit is a Power Disconnect)
- A Cab Mounted Emergency stop button within reach of the operator
- Two holding brakes
- Overspeed Sensor/Circuit
- No floor mounted emergency stop buttons
- No Overload Sensor/Circuit
- Event Frequency Calculation used to show the likelihood of an SSE earthquake is sufficiently low while the lift is being performed

A general description of the lift is as follows:

> Initially, in the first several feet of movement, directly above the flange, the major concern is whether the load has hung-up. If there is to be a hang-up, it is most likely to be observed in the first movement of the head. As shown in Table 4 checks are put into place to determine if load hang-up is occurring. Once the initial lift has been completed and it is known that the head is physically off the flange, then the lift continues vertically until the guide studs are cleared. During this stage there is concern over hang-up and binding of the head and the guide studs. After the guide studs are cleared, the head can be raised to an elevation that will allow movement to the head stand. The head could also be moved laterally away from the reactor vessel once the guide studs are cleared and then the raising completed. The final concern is the potential of a two-block event at the high point of the lift. The redundant upper limit switches are in place to prevent this event.
>
> Reinstallation of the head first presents the two-block concern as the head is lifted off the stand. Once the head has been moved back over the vessel, then the potential for load binding develops as the head is lowered to the guide studs. Once again, Table 4 indicates the measures in-place to protect the lift from this event.

The completed lift plan was documented in accordance with station procedures.

Table 3 Example of a NUREG 0554 Comparison using the NEI Guidance Document

NUREG 0554 REQUIREMENT	RESPONSE	BASIS/ACTIONS REQUIRED
2.4 Material Properties		
2.4.1 The crane and lifting fixtures for cranes already fabricated or operating may be subjected to a cold-proof test consisting of a single dummy load test.	Crane was not been cold proof tested and no fracture toughness properties are known. *In lieu of a cold-proof test, a minimum operating temperature of 70° is established per the guidance document for the Heavy Loads Initiative, NEI 08-05. WBN Procedure MI-68.001 will be revised to require verification of ambient air temperature in Upper Containment of above 70°F prior to the performance of the MCL.*	NEI Guidance Document 08-05 MI-68.001
Several sections omitted for simplicity		
4. Hoisting Machinery 4.1 Reeving System		
4.1.1 Design of the rope reeving system(s) should be dual with each system providing separately the load balance on the head and load blocks through the configuration of ropes and rope equalizer(s).	Dual reeving and equalizing systems are used for the main and auxiliary hoists. Load balancing through cross-reeving is achieved for both hoists.	Contract 75K38-86129 TVA Specification 2212, WB-DC-20-4
4.1.2 The maximum load (including static and inertia forces) on each individual wire rope in the dual reeving system with the MCL attached should not exceed 10% of the manufacturer's published breaking strength.	The factor of safety for the wire rope on the main hook is 8.57:1 for the MCL of 160 tons. This does not meet the minimum factor of safety of 10. *Based upon the guidance document for the Heavy Loads Initiative, NEI 08-05, for a hoist equipped with dual wire ropes, the minimum factor of safety must be 5:1. Therefore, 8.57:1 is considered equivalent.*	NEI Guidance Document 08-05 Supplemental Calculations (Attachment B)

Table 4 – Example of Performance of a Reactor Head Lift with an Equivalent Single-Failure-Proof Crane

Crane Action	Safety Issue	Check/Control
Head Removal		
Pre-Lift Activities		• Pre-Operational Inspection Performed at the start of refueling outage • Maintenance Rule (a)(4) measures addressed in Outage Safety Plan • Pre-Job Brief performed that includes identification of Supervisory Oversight, acceptable travel limits of crane, and establishment of Lift Management Protocol for entire lift • Backup Emergency Stop Signal such as an air horn (pre-tested) provided in case of loss of direct communication • Second Crane Operator placed in cab of crane, or the most effective location, to act as an observer/ spotter for the duration of the lift • 3-Way Direct Communications between Crane Operator, Person-in-Charge, and Signal Person via headsets (batteries refreshed in all radios) established and maintained for entire duration of lift • Load Cell check out complete with Individual to monitor load cell in place • Verified ambient air temperature is greater than 70° F • Pre-lift inspection performed just prior to lift
Initial Lift - From Flange to 24" above Flange	• Operator Error • Load Hang-Up	• Load Cell monitored (Person monitoring load cell is equipped with emergency stop signal) • 5 minute hold of the load performed after the initial lift is made to verify brakes and wire rope • Weight of head checked against station procedures • Perform visual inspections once the head is free to ensure only the head is being lifted
Raise Head Until Clear of Guide Studs - 168" to 240" above Flange	• Operator Error • Load Hang-up	• Verify load moving • Continue monitoring of load cell • Raise head at a minimum slow speed[1] • Individuals stationed to observe for head binding on guide studs •
Complete Upward Movement	• Operator Error • Two-Block Event	• Two Upper Limit Switches (Second Upper Limit shall be a Power Disconnect) • Master Switches with Spring Return to Off Feature • Two Holding Brakes
Translate to Stand	• Operator Error • Trolley Brake Failure • Bridge Brake Failure	• Trolley and Bridge movements maintained within safe load paths • Trolley and Bridge speeds minimized to eliminate load swings
Lower to Stand	• Operator Error • Hoist Brake Failure	• Two Holding Brakes
Head Installation		

Pre-Lift Activities		• Pre-Job Brief performed that includes identification of Supervisory Oversight, acceptable travel limits of crane, and establishment of Lift Management Protocol for entire lift • Backup Emergency Stop Signal such as an air horn (pre-tested) provided in case of loss of direct communication • Second Crane Operator placed in cab of crane, or the most effective location, to act as an observer/ spotter for the duration of the lift • 3-Way Direct Communications between Crane Operator, Person-in-Charge, and Signal Person via headsets (batteries refreshed in all radios) established and maintained for entire duration of lift • Load Cell check out complete with Individual to monitor load cell in place • Verified ambient air temperature is greater than 70° F • Pre-lift inspection performed just prior to lift
Initial Lift	• Operator Error • Hoist Brake Failure	• Load Cell monitored (Person monitoring load cell is equipped with emergency stop signal) • 5 minute hold of the load performed after the initial lift is made to verify brakes and wire rope • Two Holding Brakes
Translate to Vessel	• Operator Error • Trolley Brake Failure • Bridge Brake Failure	• Trolley and Bridge movements maintained within safe load paths • Trolley and Bridge speeds minimized to eliminate load swings
Lower to Guide Studs	• Operator Error • Hoist Brake Failure	• Personnel placed in key locations to perform visual observations • Two Holding Brakes • Perform close observation of the head orientation because as the wire rope reeves out, the head may tend to rotate. Slight movements in rotation of the hook, and/or bridge may be needed to compensate • Upon reaching the tapered portion of the guide studs, additional adjustments may be needed before proceeding to the straight portion of the guide stud
Lower to Flange	• Operator Error • Hoist Brake Failure • Load Hang-Up	• Two Holding Brakes to control lowering • Check that a near equal gap exists around all the guide studs • No Trolley or Bridge movements allowed once aligned with the guide studs • Load Cell monitored (Person monitoring load cell is equipped with emergency stop signal)

[1] Minimum slow speed is defined as a speed that does not create overheating of the motors or other challenges to the drive system. The speed should be as slow as is practical.

3.6 USE OF THE EQUIVALENCE APPROACH

Establishment of a single-failure-proof crane or the equivalent of a single-failure-proof crane can be performed using one of two methods. As described earlier, these methods are:

Point-by-point comparison - This may have been performed as part of a station's Phase II submittal and may have resulted in the identification of several gaps. The station could choose to do modifications to close those gaps or use the equivalence measures outlined in this guidance document as shown in Table 3. In either case, documentation should be developed in accordance with station procedures. This documentation would be available for inspection upon request.

Equivalence Evaluation - If the point-by-point comparison was not made by the station, then the station may choose to perform an equivalence evaluation. This would require the station to review the design of the crane to determine if the minimum hardware requirements and documentation requirements listed earlier have been provided. If the crane does have the minimum hardware and documentation requirements, then the station should develop a Lift Plan similar to Table 4 to describe the additional measures to be taken during the head lift. These additional measures must be based upon the guidance of Table 2. The design features and the Lift Plan must also be documented in accordance with station procedures. This documentation would be available for inspection upon request.

3.7 CONCLUSION

The use of a single-failure-proof or equivalent single-failure-proof crane provides a cost-effective alternative to performing the load drop analysis needed to meet the Heavy Loads Initiative. Most of the measures outlined in the guidance document are already performed by stations when the reactor head lifts are made. Any additional measures or modifications are expected to have minimal cost and schedule impact. The result is a reactor head lift that is performed safely and efficiently.

3.8 REFERENCES

NRC Staff Report NUREG 0554, Single-Failure-Proof Cranes for Nuclear Power Plants

NRC Staff Report NUREG 0612, Control of Heavy Loads at Nuclear Power Plants

NRC Letter Dated December 22, 1980, Control of Heavy Loads (Later identified as GL 80-113)

NRC Generic Letter 81-07, Dated February 3, 1981, Control of Heavy Loads

NRC Generic Letter 85-11, Dated June 28, 1985, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612

Crane Manufacturers Association of America Specification #70, 1975 Edition

American Society of Mechanical Engineers Specification NOG – 1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), 2004 Edition

American Society of Mechanical Engineers Specification B30.2, Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist), 1976 and 2005 Editions

4 FSAR UPDATE

4.1 INTRODUCTION

The industry initiative on Heavy Load Lifts was documented in a letter from Anthony R. Pietrangelo, Vice President, Regulatory Affairs, NEI, to James E. Dyer, Director, Office of Nuclear Reactor Regulation, US NRC, on September 14, 2007. Part of the initiative includes a requirement for an FSAR update related to the initiative. This section provides guidance on developing the FSAR update. In addition, NEI 98-03, Guidelines for Updating Final Safety Analysis Reports, Revision 1, June, 1999 should be consulted.

Regarding FSAR updates, the industry initiative on heavy load lifts (section B.4) states:

> "B. <u>For all plants with an outage beginning after July 1, 2008 and thereafter</u>:
>
> "4) In your next FSAR update, provide a summary description of your 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 FSAR."

In its Enforcement Guidance Memorandum 07-006, Enforcement Discretion for Heavy Load Handing Activities, the NRC also included the need to provide a summary description of the basis for conducting safe heavy load movements.

4.2 GUIDANCE

The following general guidance should be considered by a licensee in preparing its UFSAR update to address the control of heavy loads:

- Identify all current information related to control of heavy loads in the UFSAR (if any).
- Determine the appropriate location of this new information within the UFSAR.
- Determine the level of detail that is consistent with the current UFSAR.
- Ensure that the level of detail does not unnecessarily limit plant operation.
- Use descriptive references (such as "maintained in Maintenance procedures…").
- Ensure that previous requirements (such as commitments, license conditions, etc.) are not eliminated, reduced, or revised by this UFSAR update, unless deemed appropriate and necessary.
- Review all previously docketed correspondence on this topic and create appropriate UFSAR references. General References (see NEI 98-03, Guidelines for Updating Final Safety Analysis Reports*)* are appropriate to maintain licensee control for the operating and maintenance procedures that support the heavy loads program.
- Ensure the content meets the requirements of 10 CFR 50.71(e).

- Ensure that there is no ambiguity as to whether any procedure change can be made without NRC approval under 10 CFR 50.59.

4.3 Recommended Format and Content

The following format and content are recommended:

X. Control of Heavy Loads

X.1 Introduction/Licensing Background

Summarize the site specific licensing correspondence to and from the NRC concerning the control of heavy loads (e.g., NUREG 0612, GL 80-113, GL 81-07, GL 85-11, Bulletin 96-02, RIS 2005-25, RIS 2005-25 Supplement 1). Use General References (see NEI 98-03) to point to licensee-controlled documents that provide detail for the control of heavy loads.

X.2 Safety Basis

Describe the safety basis that ensures that the risk associated with load-handling failures is acceptably low, based on: (1) meeting the phase 1 requirements of NUREG 0612, Section 5.1.1, and (2) EITHER the use of a single failure proof crane (or equivalent for the reactor vessel head lift), OR a load drop analysis that demonstrates the fuel remains covered and cooled.

X.3 Scope of Heavy Load Handling Systems

Based on the licensee’s response to Phase I, list or describe the load handling equipment that is in the scope of NUREG 0612, Section 5.1.1.

From NUREG 0612, the scope of cranes includes:

> “All plants have overhead handling systems that are used to handle heavy loads in the area of the reactor vessel or spent fuel in the spent fuel pool. Additionally, loads may be handled in other areas where their accidental drop may damage safe shutdown systems….”

The list can be presented in tabular form, or provide a General Reference to letters and documents that describe the overhead handling systems. The level of detail should be determined by the licensee and be consistent with the rest of the UFSAR.

X.4 Control of Heavy Loads Program

Outline of suggested text for introduction to this section:

The Control of Heavy Loads Program consists of the following:
1. *Licensee commitments in response to NUREG-0612, Phase I elements*
2. *For RPVH lifts*
    i. *EITHER a load drop analysis with assumptions (lift height, load weight, medium present) from the head drop analysis incorporated into plant procedures.*
    ii. *OR single failure proof crane (or equivalent) with justification.*
3. *For spent fuel cask lifts over the spent fuel pool, either a load drop analysis OR single failure proof crane.*

X.4.1 [Licensee] Commitments in Response to NUREG 0612, Phase I Elements

For cranes that are within the scope of NUREG 0612, seven elements must be met as described in NUREG 0612, Section 5.1.1 commonly known as Phase I ["…Accordingly, all plants should satisfy each of the following for handling heavy loads that could be brought in proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel pool area or containment (PWRs), in the reactor building (BWRs), and in other plant areas."]

List the seven elements of Phase I and briefly describe how they are being implemented:

1. Definition of safe load paths
2. Development of load handling procedures
3. Qualifications, training, and specified conduct of crane operators
4. Special lifting devices should satisfy the guidelines of American National Standards Institute (ANSI) N14.6-1978
5. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9
6. Periodic inspection and testing of cranes
7. Design of cranes to ANSI B30.2 or CMAA-70

X.4.2 Reactor Pressure Vessel Head (RPVH) Lifting Procedures

If a load drop analysis is being used to support lifts of the RPVH, describe the assumptions (restrictions on load height, load weight, and medium present under the load) from the head drop analysis that are incorporated into plant procedures.

Suggested text:

> *To control Reactor Pressure Vessel Head lifts, [plant] procedures are used to control the lift and replacement of the reactor pressure vessel head. These procedures establish*

limits on load height, load weight, and medium present under the load. These procedures: (1) use the guidance and acceptance criteria in NEI 08-05 Industry Initiative on Control of Heavy Loads [state references to the analysis]; and (2) provide additional assurance that the core will remain covered and cooled in the event of a postulated reactor pressure vessel head drop.

If a single failure proof crane or equivalent is being used to support lifts of the RPVH, describe the design elements needed to make the single failure proof crane or equivalent description complete and accurate.

X.4.3 Single Failure Proof Cranes for Spent Fuel Casks

For the spent fuel casks, either describe the design elements needed to make the single failure proof crane description complete and accurate, or describe the assumptions used in the spent fuel cask drop analysis. NOTE: In some FSARs, the spent fuel cask crane may be covered in a different section.

X.5 Safety Evaluation

This is the conclusion section. It should provide a clear basis for the site's conclusion that heavy load lifts are done safely.

- Controls implemented by NUREG 0612 Phase 1 elements make the risk of a load drop very unlikely.

  AND

- In the event of a postulated load drop, the consequences are acceptable, as demonstrated by the load drop analysis. Restrictions on load height, load weight, and medium under the load are reflected in plant procedures.

  OR
- The use of a single failure proof crane or equivalent makes the risk of a load drop extremely unlikely and acceptably low.

  AND

- The risk associated with the movement of heavy loads is evaluated and controlled by station procedures.

APPENDIX: CORRESPONDENCE WITH NRC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 16, 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 staff has completed its evaluation of the guidance documents entitled "Industry Criteria for Reactor Vessel Head Load Drop and Consequence Analysis" and "Reactor Head Lift Single Failure Proof Crane Equivalence," which you provided as enclosures to letters dated April 17, 2008, and April 22, 2008, respectively. These documents are in the Agencywide Documents Access and Management System (ADAMS) as Accession Nos. ML081300340 and ML081300045.

The NRC staff's understanding of these guidance documents has benefited from public meetings held on December 13, 2007, February 1, 2008, April 8, 2008, and April 17, 2008. I am writing to inform you of the staff's position regarding endorsement of these guidelines as approaches acceptable to the NRC staff for implementation of the industry initiative and changes to plant licensing bases.

The guidance contained in the "Industry Criteria for Reactor Vessel Head Load Drop and Consequence Analysis" addresses: (1) general requirements for the analysis, (2) selection of material properties, (3) analytical modeling requirements, and (4) acceptance criteria when evaluating the effects of postulated heavy load drops. The modeling requirements and acceptance criteria contain guidelines that apply to both stress-based discrete component models and strain-based finite element analyses. The staff understands and appreciates that NEI has updated the guidance document in response to NRC staff concerns discussed during the previous public meetings. These concerns related to the uncertainty in the process used to establish material properties and the adequacy of the margin associated with the proposed acceptance criteria to compensate for uncertainty associated with the selected analytical methods.

With regard to the establishment of appropriate material properties, the earlier draft industry guidelines specify that actual test data (e.g., certified material test reports (CMTRs)) may be used to establish material properties. During previous public meetings, the staff expressed concern that the CMTRs may provide unrealistically high values for these 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 considers application of this revised guidance acceptable for establishment of material properties.

The staff considers the American Society of Mechanical Engineers' Boiler and Pressure Vessel Code, Section III, Appendix F acceptance criteria for limiting events (i.e., Service Level D) acceptable for the analytical methods proposed in the draft guidance. For energy balance evaluations using the large-displacement finite element methods described in the guidance, the staff finds 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, NEI has proposed more relaxed strain-based criteria for application to large-displacement finite element evaluations of coolant retaining components and component supports. The staff will review for acceptance the proposed NEI strain-based criteria, if such criteria is developed in conjunction with accurate benchmarking to large displacement tests of similar material in similar configurations. Under such an approach, strain-based acceptance criteria proposed by NEI may be acceptable as a result of the reduced uncertainty provided by the benchmarking.

The staff finds the guidance contained in the "Reactor Head Lift Single Failure Proof Crane Equivalence" 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 is acceptable.

We believe reactor vessel head lifts beginning after July 1, 2008, can readily be performed consistent with appropriate load drop analyses or with cranes satisfying the guidelines described above for single failure proof crane equivalence. The 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 would perform these interim analyses for reactor vessel head lifts over flooded refueling cavities that conform to the proposed analysis guidelines that we endorsed above. The staff expects that the margin provided by the water cushion would allow for timely completion of these analyses.

The NRC staff appreciates your continued interest in implementation of your initiative on heavy load handling. We intend to follow this letter with formal endorsement of the NEI guidance as indicated above through a generic communication and an eventual update to the Standard Review Plan, NUREG-0800. The staff will provide opportunity for public comment in association with development of the RIS. Please contact Mr. Steven Jones of the NRC staff at (301) 415-2712 to discuss any other concerns.

Sincerely,

William H Ruland

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

PROJ: 689

cc: See Next Page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 27, 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:

I am writing to clarify the approaches the Nuclear Regulatory Commission (NRC) staff considers acceptable for interim analyses, which I described in my letter to you dated May 16, 2008. In a telephone conversation on May 20, 2008, you indicated to me that the need to evaluate the reactor vessel supports and attached piping in completing interim analyses for reactor vessel head lifts over flooded refueling cavities was unclear.

The 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. As I stated in my letter dated May 16, 2008, licensees may perform these interim analyses for reactor vessel head lifts over flooded refueling cavities that conform to the proposed analysis guidelines contained in "Industry Criteria for Reactor Vessel Head Load Drop and Consequence Analysis," which is in the NRC Agencywide Documents Access and Management System (ADAMS) as Accession No. ML081300340.

Many existing analyses completed since the late 1970s generally conform to these guidelines, with the exception that component supports and attached piping were not explicitly considered. Instead, the analyses assumed the reactor vessel supports were rigid, which is bounding with respect to stresses developed in the reactor vessel near the supports. These analyses considered form drag as the head moved through water, but neglected surface and hydraulic effects that are certain to further reduce the impact forces. The staff considers the margin provided by the neglected surface and hydraulic effects adequate to ensure reactor vessel support and loop piping stresses would remain within allowable limits. Therefore, these analyses are acceptable for use as the basis for comparative analyses when the refueling cavity will be flooded (i.e., the lower of fully flooded or flooded to within 15 feet of the bottom of the reactor vessel head whenever the head is above the guide studs) during the reactor vessel head lift. These comparative (parametric) analyses for lifts over flooded refueling cavities need only consider differences in reactor head lift height, head assembly weight, reactor vessel material properties, and reactor vessel and head geometry. However, the NRC staff understands these analyses will only be used on an interim basis during the period necessary to either complete more detailed load drop analyses using energy balance methods or enhance the reliability of the handling system to single failure proof crane equivalence.

From your letter dated April 22, 2008, we understand that you will combine into a single industry guideline the guidance documents related to load drop analyses and single failure proof crane equivalence with guidance on updating the plant Final Safety Analysis Report (FSAR) to reflect the current licensing basis with respect to control of heavy loads. We consider this latter the current licensing basis with respect to control of heavy loads. We consider this latter guidance essential to support clear updates to the FSAR and a more consistent licensing basis. For planning purposes in support of potential NRC endorsement through a Regulatory Issue Summary, please inform us of your schedule for completion of the combined industry guideline.

The NRC staff appreciates your continued interest in implementation of your initiative on heavy load handling. Please contact Mr. Steven Jones of the NRC staff at (301) 415-2712 to discuss any other concerns.

Sincerely,

William H. Ruland

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

PROJ: 689

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