INDUSTRY CRITERIA FOR REACTOR VESSEL LOAD DROP AND CONSEQUENCE ANALYSIS

On September 14, 2007, the industry's Nuclear Strategic Initiatives 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. Enclosure 1 is a copy of the letter transmitting the industry initiative to the NRC.

An industry task force on heavy loads was established to develop guidelines for various aspects of the initiative. This section of the guidelines addresses reactor vessel load drop analysis. Additional sections will be developed to address other aspects of the initiative.

A subgroup of industry specialists in load drop evaluations was formed to develop this section of the guidelines. The group is 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.

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. Appendix A provides a technical basis for the guidelines.

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	N/A

handling of the heavy load being analyzed, or whenever the building negative pressure rises above (-)1/8 inch (-3 m) water gauge	
(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

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.

1.0 General Requirements for Analysis

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

1.2 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.

1.3 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.

1.4 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.

2.0 Material Requirements

2.1 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.

2.2 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.

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

2.4 A Dynamic Increase Factor (DIF) due to dynamic strain rate effects can be applied to the static stress strain diagram with appropriate technical justification.

3.0 Modeling Requirements

3.1 The reactor vessel head is not required to be modeled as a rigid mass, provided that the head and any associated structures are explicitly accounted for in the model. The analysis may account for the stiffness of the head by appropriate analysis methods. However, if the head is not modeled explicitly, the head must be modeled as a rigid mass.

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

3.3 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.

3.4 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.

3.5 The use of a coefficient of restitution shall not be used for an elastic-plastic analysis in which the head and/or flange are modeled explicitly. For discrete-mass models of the head and reactor vessel, any energy loss due to impact, caused by the selection of the coefficient of restitution and the magnitude of the impacted mass, must be justified if used in the analysis.

3.6 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. Higher damping values may be used with an appropriate technical justification. The foundation mass and radiation damping may also be included.

3.7 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.

3.8 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.

4.0 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.

4.1 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.

4.1.1 Pressure 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.

4.1.2 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.

4.2 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 of 4.1. Strain criteria are only applicable to materials with Sy/Su < 0.7.

4.2.1 Strain Acceptance Criteria for Coolant Retaining Components

- Average (through thickness) equivalent total strain is limited to strain of 0.5 ε_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 ε_u.

4.2.2 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 $0.5\varepsilon_u$ where ε_u is the strain at ultimate stress (see Figure 1).
- Tensile and bending – members loaded in bending shall limit the maximum average tensile strain plus linearized (through thickness) equivalent bending strain to $1.0\varepsilon_u$.
- 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 $0.6S_u$.

Welded Structures

- Full penetration structural welds shall be considered equivalent to the base 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.

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 349-01 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 349-01 will be met with a capacity reduction factor of 0.9.

4.3 Maximum Deflection

4.3.1 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.

4.3.2 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.

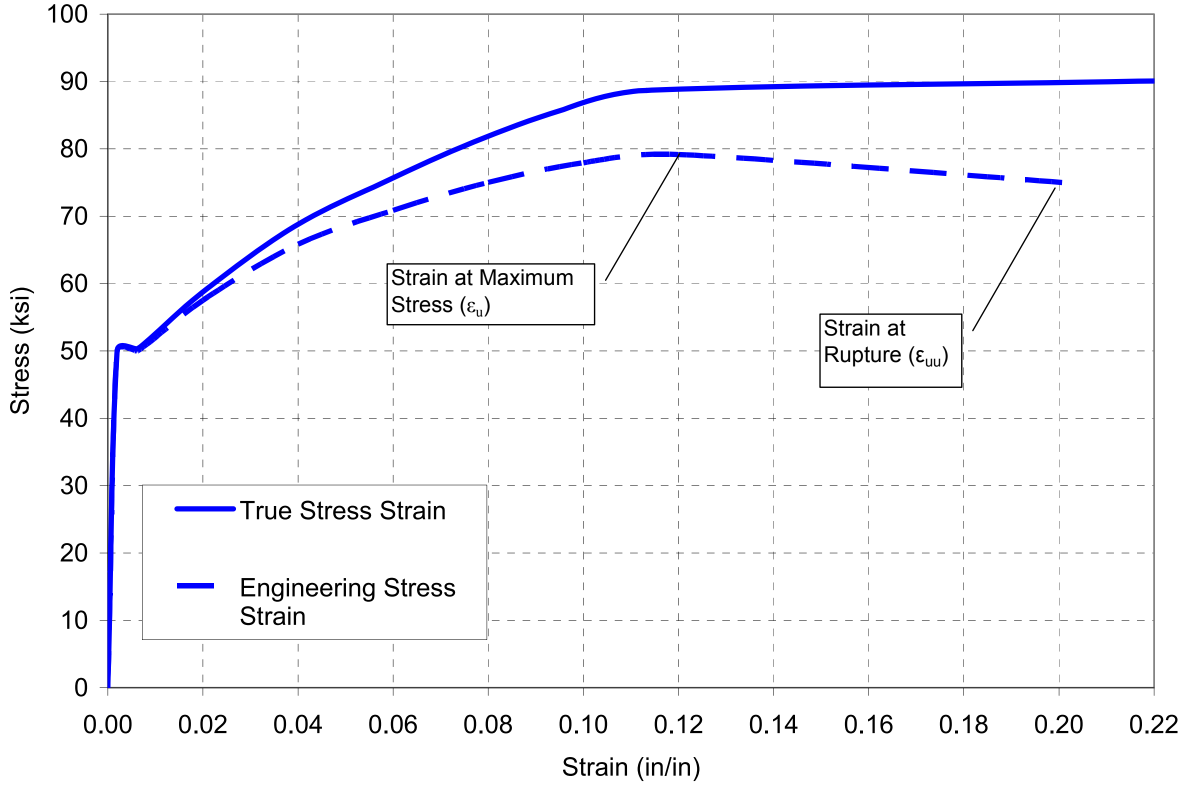

Figure 1. Definitions of Strains

4.4 Determination of Material Properties

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

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

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

4.5 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

4.6 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)

Appendix A

TECHNICAL BASIS DOCUMENT

A.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.

A.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.

A.3 Modeling Requirements

NUREG-0612 requires 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.

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.

NUREG-0612 prohibits 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.

The target critical damping values are consistent with the damping for welded steel and concrete structures for SSE events.

NUREG-0612 requires 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 (steady-state).

A.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 criteria 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 Divison 1, Appendix F (30% and 10% as discussed above). This indicates that the strain criteria is 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)

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Mr. James E. Dyer
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Washington, DC 20555-0001

Subject: Industry Initiative on Heavy Load Lifts

Project Number: 689

Dear Mr. Dyer:

Over the last several weeks, we have discussed with you and members of the NRC staff concerns regarding the interpretation and implementation of regulatory guidance associated with heavy load lifts. While there have been no significant events associated with heavy load lifts, we have identified a lack of consistency in plant licensing bases that pertain to this issue. Therefore, the industry has approved a formal initiative that 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 is enclosed for your information.

NEI will be forming a task force to assist industry implementation of the initiative. We intend to continue to communicate with the NRC staff in this effort to ensure that all concerns are appropriately addressed and that the initiative achieves its intended outcome.

Please contact me if you have any questions regarding the initiative.

Sincerely,

Anthony R. Pietrangelo

Anthony R. Pietrangelo

Enclosure

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ENCLOSURE

INDUSTRY INITIATIVE ON HEAVY LOAD LIFTS

A. For plants with an outage beginning before July 1, 2008:

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 not 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. For all plants with an outage beginning after July 1, 2008 and thereafter:

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.