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

June 26, 1980

Docket No. 50-289

Mr. R. C. Arnold
Senior Vice President
Metropolitan Edison Company
100 Interpace Parkway
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Dear Mr. Arnold:

In January 1978, the NRC published NUREG-0410 entitled, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants - Report to Congress". As part of this program, the Task Action Plan for Unresolved Safety Issue Task No. A-36, "Control of Heavy Loads Near Spent Fuel," was issued.

We have completed our review of load handling operations at nuclear power plants. A report describing the results of this review will be issued in the near future as NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants - Resolution of TAP A-36." This report contains several recommendations to be implemented by all licensees to assure the safe handling of heavy loads.

At the Indian Point Units 2 and 3, Zion Units 1 and 2, and Three Mile Island Unit 1 facilities, we are requesting licensee action to begin to implement these recommendations at this time on the schedule indicated in this letter.

To expedite your compliance with this request, we have enclosed the following:

1. Guidelines for Control of Heavy Loads (Enclosure 1).

2. Staff Position - Interim Actions for Control of Heavy Loads (Enclosure 2).

3. Request for Additional Information on Control of Heavy Loads (Enclosure 3).

You are requested to review your controls for the handling of heavy loads to determine the extent to which the guidelines of Enclosure 1 are presently satisfied at your facility, and to identify the required changes and modifications in order to fully satisfy these guidelines.

You are requested to implement the interim actions described in Enclosure 2 as soon as possible but no later than 90 days from the date of this letter.

You are further requested to submit a report documenting the results of your review and the required changes and modifications. This report should include the information identified in Sections 2.1 through 2.4 of Enclosure 3, on how the guidelines of NUREG-0612 will be satisfied. This report should be submitted not later than the following schedule.

- Submit the Section 2.1 information within three months from the date of this letter.

- Submit the Sections 2.2, 2.3, and 2.4 information within six months.

You should commence implementation of required changes and modifications as soon as possible without waiting on staff review, with the objective of completing all procedural and documentation changes, beyond the above interim actions, within two years of submittal of Section 2.4 for the above report.

Please notify your assigned NRC Project Manager if you will not be able to maintain these schedules.

Sincerely,

Darrell G. Eisenhut, Director
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As stated

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5. GUIDELINES FOR CONTROL OF HEAVY LOADS

Our evaluation of the information provided by licensees indicates that existing measures at operating plants to control the handling of heavy loads cover certain of the potential problem areas, but do not adequately cover the major causes of load handling accidents. These major causes include operator errors, rigging failures, lack of adequate inspection and inadequate procedures. The measures in effect vary from plant to plant, with some having detailed procedures while others do not, some have performed analyses of certain postulated load drops, certain plants have single-failure-proof cranes, some PWR's have rapid containment isolation on high radiation, and many plants have technical specifications that prohibit handling of heavy loads or a spent fuel cask over the spent fuel pool. To provide adequate measures that minimize the occurrence of the principal causes of load handling accidents and to provide an adequate level of defense-in-depth for handling of heavy loads near spent fuel and safe shutdown systems, the measures in effect should be upgraded.

5.1 Recommended Guidelines

The following sections describe various alternative approaches which provide acceptable measures for the control of heavy loads. The objectives of these guidelines are to assure that either (1) the potential for a load drop is extremely small, or (2) for each area addressed, the following evaluation criteria are satisfied:

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);

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 k_{eff} 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.

After reviewing the historical data available on crane operations, identifying the principal causes of load drops, and considering the type and frequency of load handling operations at nuclear power plants, the NRC staff has developed an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads. This philosophy encompasses an intent to prevent as well as mitigate the consequences of postulated accidental load drops. The following summarizes this defense-in-depth approach:

(1) Provide sufficient operator training, handling system design, load handling instructions, and equipment inspection to assure reliable operation of the handling system; and

(2) Define safe load travel paths through procedures and operator training so that to the extent practical heavy loads avoid being carried over or near irradiated fuel or safe shutdown equipment; and

(3) Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Certain alternative measures may be taken to compensate for deficiencies in (2) and (3) above, such as the inability to prevent a particular heavy load from being brought over spent fuel (e.g., reactor vessel head). These alternative measures can include: increasing crane reliability by providing dual load paths for certain components, increased safety factors, and increased inspection as discussed in Section 5.1.6 of this report; restricting crane operations in the spent fuel pool area (PWRs) until fuel has decayed so that off-site releases would be sufficiently low if fuel were damaged; or analyzing the effects of postulated load drops to show that consequences are within acceptable limits. Even if one of these alternative measures is selected, (1) and (2) above should still be satisfied to provide maximum practical defense-in-depth.

The following sections provide guidelines on how the above defense-in-depth approach may be satisfied for various plant areas. Fault trees and associated probabilities were developed and used as described in Bases for Guidelines, Section 5.2 of this report, to evaluate the adequacy of these guidelines and to assure a consistent level of protection for the various areas.

5.1.1 General

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. 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 and in containment (PWRs), in the reactor building (BWRs), and in other plant areas.

(1) Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy leads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members. beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee.

(2) Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of this report. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe load path; and other special precautions.

(3) Crane operators should be trained, qualified and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, "Overhead and Gantry Cranes."

(4) Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials." This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used.* This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device.

(5) Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings." However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load.* The rating identified on the sling should be in terms of the "static load" which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used.

(6) The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, "Overhead and Gantry Cranes," with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, tests, and maintenance should be performed prior to their use.)

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*For the purpose of selecting the proper sling, loads imposed by the SSE need not be included in the dynamic loads imposed on the sling or lifting device.

(7) The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, "Overhead and Gantry Cranes" and of CMAA-70, "Specifications for Electric Overhead Travelling Cranes." An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied.

5.1.2 Spent Fuel Pool Area - PWR

Many PWR's require that the spent fuel shipping cask be placed in the spent fuel pool for loading. Additionally, other heavy loads may be carried over or near the spent fuel pool using the overhead crane, including plant equipment, rad-waste shipping casks, the damaged fuel container and replacement fuel storage racks. Additionally, certain crane failures could cause the crane lower load block to be dropped, and therefore this should also be considered as a heavy load. The fuel handling crane is used for moving fuel and is generally not used for handling of heavy loads. To provide assurance that the evaluation criteria of Section 5.1 are met for load handling operations in the spent fuel pool area, in addition to satisfying the general guidelines of Section 5.1.1, one of the following should be satisfied:

(1) The overhead crane and associated lifting devices used for handling heavy loads in the spent fuel pool area should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

OR

(2) Each of the following is provided:

(a) Mechanical stops or electrical interlocks should be provided that prevent movement of the overhead crane load block over or within 15 feet horizontal (4.5 meters) of the spent fuel pool. These mechanical stops or electrical interlocks should not be bypassed when the pool contains "hot" spent fuel, and should not be bypassed without approval from the shift supervisor (or other designated plant management personnel). The mechanical stops and electrical interlocks should be verified to be in place and operational prior to placing "hot" spent fuel in the pool.

(b) The mechanical stops or electrical interlocks of 5.1.2(2)(a) above should also not be bypassed unless an analysis has demonstrated that damage due to postulated load drops would not result in criticality or cause leakage that could uncover the fuel.

(c) To preclude rolling if dropped, the cask should not be carried at a height higher than necessary and in no case more than six (6) inches (15 cm) above the operating floor level of the refueling building or other components and structures along the path of travel.

(d) Mechanical stops or electrical interlocks should be provided to preclude crane travel from areas where a postulated load drop could damage equipment from redundant or alternate safe shutdown paths.

(e) Analyses should conform to the guidelines of Appendix A.

OR

(3) Each of the following are provided (Note: This alternative is simlar to (a) above, except it allows movement of a heavy load, such as a cask, into the pool while it contains "hot" spent fuel if the pool is large enough to maintain wide separation between the load and the "hot" spent fuel.):

(a) "Hot" spent fuel should be concentrated in one location in the spent fuel pool that is separated as much as possible from load paths.

(b) Mechanical stops or electrical interlocks should be provided to prevent movement of the overhead crane load block over or within 25 feet horizontal (7.5 m) of the "hot" spent fuel. To the extent practical, loads should be moved over load paths that avoid the spent fuel pool and kept at least 25 feet (7.5 m) from the "hot" spent fuel unless necessary. When it is necessary to bring loads within 25 feet of the restricted region, these mechanical stops or electrical interlocks should not be bypassed unless the spent fuel has decayed sufficiently as shown in Table 2.1-1 and 2.1-2, or unless the total inventory of gap activity for fuel within the protected area would result in offsite doses less than ¼ of 10 CFR Part 100 if released, and such bypassing should require the approval from the shift supervisor (or other designated plant management individual). The mechanical stops or electrical interlocks should be verified to be in place and operational prior to placing "hot" spent fuel in the pool.

(c) Mechanical stops or electrical interlocks should be provided to restrict crane travel from areas where a postulated load drop could damage equipment from redundant or alternate safe shutdown paths. Analyses have demonstrated that a postulated load drop in any location not restricted by electrical interlocks or mechanical stops would not cause damage that could result in criticality, cause leakage that could uncover the fuel, or cause loss of safe shutdown equipment.

(d) To preclude rolling, if dropped, the cask should not be carried at a height higher than necessary and in no case more than six (6) inches (15 cm) above the operating floor level of the refueling building or other components and structures along the path of travel.

(e) Analyses should conform to the guidelines of Appendix A.

OR

(4) The effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1 of this report. These analyses should conform to the guidelines of Appendix A.

5.1.3 Containment Building - PWR

PWR containment buildings contain a polar crane that is used for removing and reinstalling shield plugs, the reactor vessel head, upper vessel internals, and on occasion, other heavy equipment such as the reactor coolant pump, the reactor vessel inspection platform, and the cask used for damaged fuel. Additionally the crane load block may be moved over fuel in the reactor when handling smaller loads or no load at all. Due to the weight of the load block alone, this should also be considered as a heavy load. To provide assurance that the criteria of Section 5.1 are met for load handling operations in the containment building, in addition to satisfying the general guidelines of Section 5.1.1, one of the following should be satisfied:

(1) The crane and associated lifting devices used for handling heavy loads in the containment building should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

OR

(2) Rapid containment isolation is provided with prompt automatic actuation on high radiation so that postulated releases are within limits of evaluation Criterion I of Section 5.1 taking into account delay times in detection and actuation; and analyses have been performed to show that evaluation criteria II, III, and IV of Section 5.1 are satisfied for postulated load drops in this area. These analyses should conform to the guidelines of Appendix A.

OR

(3) The effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1. Loads analyzed should include the following: reactor vessel head; upper vessel internals; vessel inspection platform; cask for damaged fuel; irradiated sample cask; reactor coolant pump; crane load block; and any other heavy loads brought over or near the reactor vessel or other equipment required for continued decay heat removal and maintaining shutdown. In this analysis, credit may be taken for containment isolation if such is provided; however analyses should establish adequate detection and isolation time. Additionally, the analysis should conform to the guidelines of Appendix A.

5.1.4 Reactor Building - BWR

The reactor building in BWRs typically contains the reactor vessel and spent fuel pool, as well as various safety-related equipment.

The reactor building overhead crane may be used in many day-to-day operations such as moving various shielded shipping casks or handling plant equipment related to maintenance or modification activities. The crane is also used during refueling operations for removal and reinstallation of shield plugs, drywell head, reactor vessel head, steam dryers and separators, and refueling canal plugs and gates. The crane would also be used subsequent to refueling for handling of the spent fuel shipping cask. This cask may be lifted as high as 100 feet (30 m) above the grade elevation at which the cask is brought into the reactor building. Additionally the overhead crane's load block may be moved over fuel in the reactor or over the spent fuel pool when handling smaller loads or no load at all. Due to the weight of the load block alone, this should also be considered as a heavy load.

To assure that the evaluation criteria of Section 5.1 are satisfied one of the following should be met in addition to satisfying the general guidelines of Section 5.1.1:

(1) The reactor building crane, and associated lifting devices used for handling the above heavy loads, should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

OR

(2) The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.. are satisfied. The loads analyzed should include: shield plugs, drywell head, reactor vessel head; steam dryers and separators; refueling canal plugs and gates; shielded spent fuel shipping casks; vessel inspection platform; and any other heavy loads that may be brought over or near safe shutdown equipment as well as fuel in the reactor vessel or the spent fuel pool. Credit may be taken in this analysis for operation of the Standby Gas

Treatment System if facility technical specifications require its operation during periods when the load being analyzed would be handled. The analysis should also conform to the guidelines of Appendix A.

5.1.5 Other Areas

In other plant areas, loads may be handled which, if dropped in a certain location, may damage safe shutdown equipment. Although this is not a concern at all plants, loads that may damage safe shutdown equipment at some plants include the spent fuel shipping cask, turbine generator parts in the turbine building, and plant equipment such as pumps, motors, valves, heat exchangers, and switchgear. Some of these loads may be less than the weight of a fuel assembly with its handling tool, but may be sufficient to damage safe shutdown equipment.

(1) If safe shutdown equipment are beneath or directly adjacent to a potential travel load path of overhead handling systems, (i.e., a path not restricted by limits of crane travel or by mechanical stops or electrical interlocks) one of the following should be satisfied in addition to satisfying the general guidelines of Section 5.1.1:

(a) The crane and associated lifting devices should conform to the single-failure-proof guidelines of Section 5.1.6 of this report;

OR

(b) If the load drop could impair the operation of equipment or cabling associated with redundant or dual safe shutdown paths, mechanical stops or electrical interlocks should be provided to prevent movement of loads in proximity to these redundant or dual safe shutdown equipment (In this case credit should not be taken for intervening floors unless justified by analysis).

OR

(c) The effects of load drops have been analyzed and the results indicate that damage to safe shutdown equipment would not preclude operation of sufficient equipment to achieve safe shutdown. Analyses should conform to the guidelines of Appendix A, as applicable.

(2) Where the safe shutdown equipment has a ceiling separating it from an overhead handling system, an alternative to Section 5.1.5(1) above would be to show by analysis that the largest postulated load handled by the handling system would not penetrate the ceiling or cause spalling that could cause failure of the safe shutdown equipment.

5.1.6 Single-Failure-Proof Handling Systems

For certain areas, to meet the guidelines of Sections 5.1.2, 5.1.3, 5.1.4, or 5.1.5, the alternative of upgrading the crane and lifting devices may be chosen. The purpose of the upgrading is to improve the reliability of the handling system through increased factors of safety and through redundancy or duality in certain active components. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," provides guidance for design, fabrication, installation, and testing of new cranes that are of a high reliability design. For operating plants, Appendix C to this report, "Modification of Existing Cranes," provides guidelines on implementation of NUREG-0554 for operating plants and plants under construction.

Section 5.1.1 of this report provides certain guidance on slings and special handling devices. Where the alternative is chosen of upgrading the handling system to be "single-failure-proof", then steps beyond the general guidelines of Section 5.1.1 should be taken.

Therefore, the following additional guidelines should be met where the alternative of upgrading handling system reliability is chosen:

(1) Lifting Devices:

(a) Special lifting devices that are used for heavy loads in the area where the crane is to be upgraded should meet ANSI N14.6 1978, "Standard For Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More For Nuclear Materials," as specified in Section 5.1.1(4) of this report except that the handling device should also comply with Section 6 of ANSI N14.6-1978. If only a single lifting device is provided instead of dual devices, the special lifting device should have twice the design safety factor as required to satisfy the guidelines of Section 5.1.1(4). However, loads that have been evaluated and shown to satisfy the evaluation criteria of Section 5.1 need not have lifting devices that also comply with Section 6 of ANSI N14.6.

(b) Lifting devices that are not specially designed and that are used for handling heavy loads in the area where the crane is to be upgraded should meet ANSI B30.9 - 1971, "Slings" as specified in Section 5.1.1(5) of this report, except that one of the following should also be satisfied unless the effects of a drop of the particular load have been analyzed and shown to satisfy the evaluation criteria of Section 5.1:

(i) Provide dual or redundant slings or lifting devices such that a single component failure or malfunction in the sling will not result in uncontrolled lowering of the load;

OR

(ii) In selecting the proper sling, the load used should be twice what is called for in meeting Section 5.1.1(5) of this report.

(2) New cranes should be designed to meet NUREG-0554, "Single-Failure-Proof Cranes For Nuclear Power Plants." For operating plants or plants under construction, the crane should be upgraded in accordance with the implementation guidelines of Appendix C of this report.

(3) Interfacing lift points such as lifting lugs or cask trunions should also meet one of the following for heavy loads handled in the area where the crane is to be upgraded unless the effects of a drop of the particular load have been evaluated and shown to satisfy the evaluation criteria of Section 5.1:

(a) Provide redundancy or duality such that a single lift point failure will not result in uncontrolled lowering of the load; lift points should have a design safety factor with respect to ultimate strength of five (5) times the maximum combined concurrent static and dynamic load after taking the single lift point failure.

OR

(b) A non-redundant or non-dual lift point system should have a design safety factor of ten (10) times the maximum combined concurrent static and dynamic load.

ENCLOSURE 2

STAFF POSITION - INTERIM ACTIONS FOR CONTROL OF HEAVY LOADS

(1) Safe load paths should be defined per the guidelines of Section 5.1.1(1) (See Enclosure 1);

(2) Procedures should be developed and implemented per the guidelines of Section 5.1.1(2) (See Enclosure 1);

(3) Crane operators should be trained, qualified and conduct themselves per the guidelines of Section 5.1.1(3) (See Enclosure 1);

(4) Cranes should be inspected, tested, and maintained in accordance with the guidelines of Section 5.1.1(6) (See Enclosure 1); and

(5) In addition to the above, special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (1) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (2) visual inspections of load bearing components of cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (3) appropriate repair and replacement of defective components; and (4) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operations, and content of procedures.

Enclosure (3)

REQUEST FOR ADDITIONAL INFORMATION ON CONTROL OF HEAVY LOADS

1. INTRODUCTION

Verification by the licensee that the risk associated with load-handling failures at nuclear power plants is extremely low will require a systematic evaluation of all load-handling systems at each site. The following specific information requests have been organized to support such a systematic approach, and provide a basis for the staff's review of the licensee's evaluation. Additionally, they have been organized to address separately the two hazards requiring investigation (i.e., radiological consequences of damage to fuel and unavailability consequences of damage to certain systems). The following general information is provided to assist in this evaluation and reduce the need for clarification as to the intent and expected results of this inquiry.

1. Risk reduction can be demonstrated by either of two approaches:

   a. The possibility of failure is extremely low due to handling-system design features (NUREG 0612, Section 5.1.6).

   b. The consequences of a failure can be shown to be acceptable (NUREG 0612, Section 5.1, Criteria I-IV).

   Regardless of the approach selected, the general guidelines of NUREG 0612, Section 5.1.1, should be satisfied to provide maximum practical defense-in-depth.

2. Evaluations concerning radiological consequences or criticality safety, where used, can rely on either the adoption of generic analyses reported in NUREG 0612, requiring only verfication that these generic assumptions are valid for a specific site, or employ a site-specific analysis.

3. Systems required for safe shutdown and continued decay heat removal are site-specific and are not, therefore, identified in this request. Individual plants should consider systems and components identified in Regulatory Guide 1.29, Position C.1 (except those systems or portions of systems that are required for (a) emergency core cooling, (b) post-accident containment heat removal, or (c) post-accident containment atmosphere cleanup), for evaluation and recognize that the approach taken in this respect is similar to that identified in Regulatory Guide 1.29, Position C.2. The fact that a load-handling system may be prevented from operating during plant conditions requiring the actual or potential use of some of these systems, is re-

cognized in this respect for information.

4. The scope of this systematic review should include all heavy loads carried in areas where the potential for non-compliance with the acceptance criteria (NUREG 0612, Section 5.1) exists. A summary of typical loads to be considered has been provided in Attachment 6. It is recognized that some cranes will carry additional miscellaneous loads, some of which are not identifiable in detail in advance. In such cases an evaluation or analysis demonstrating the acceptability of the handling of a range of loads should be provided.

5. At some sites loads which must be evaluated will include licensed shipping casks provided for the transportation of irradiated fuel, solidified radioactive waste, spent resins, or other byproduct material. Licensing under 10CFR71 is not evidence that lifting devices for these shipping casks meet the criteria specified in NUREG 0612, Sections 5.1.1(4), 5.1.1(5), 5.1.6(1), or 5.1.6(3), as appropriate, and thus does not eliminate the need to provide appropriate information concerning these devices. A tabulation (Attachment 7) is provided to indicate multiple-site use of these shipping casks.

The results of the licensee's evaluation, as reported in response to this request, should provide information sufficient for the staff to conduct an independent review to determine that the intent of this effore (i.e., the uniform reduction of the potential hazard from load-handling-system failures) has been satisfied.

2. INFORMATION REQUESTED FROM THE LICENSEE

2.1 GENERAL REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS

NUREG 0612, Section 5.1.1, identifies several general guidelines related to the design and operation of overhead load-handling systems in the areas where spent fuel is stored, in the vicinity of the reactor core, and in other areas of the plant where a load drop could result in damage to equipment required for safe shutdown or decay heat removal. Information provided in response to this section should identify the extent of potentially hazardous load-handling operations at a site, the extent of conformance to appropriate load-handling guidance, and the changes required in order to conform to the guidance.

1. Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any

interlocks, technical specifications, operating procedures, or detailed structural analysis).

2. Justify the exclusion of any overhead handling system from the above category by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or core decay heat removal.

3. With respect to the design and operation of heavy-load-handling systems in the containment and the spent-fuel-pool area and those load-handling systems identified in 2.1-1, above, provide your evaluation concerning compliance with the guidelines of NUREG 0612, Section 5.1.1. The following specific information should be included in your reply:

   a. Drawings or sketches sufficient to clearly identify the location of safe load paths, spent fuel, and safety-related equipment.

   b. A discussion of measures taken to ensure that load-handling operations remain within safe load paths, including procedures, if any, for deviation from these paths.

   c. A tabulation of heavy loads to be handled by each crane which includes the load identification, load weight, its designated lifting device, and verification that the handling of such load is governed by a written procedure containing, as a minimum, the information identified in NUREG 0612, Section 5.1.1(2).

   d. Verification that lifting devices identified in 2.1.3-c, above, comply with the requirements of ANSI 14.6-1978, or ANSI B20.9-1971 as appropriate. For lifting devices where these standards, as supplemented by NUREG 0612, Section 5.1.1(4) or 5.1.1(5), are not met, describe any proposed alternatives and demonstrate their equivalency in terms of load-handling reliability.

   e. Verification that ANSI B30.2-1976, Chapter 2-2, has been invoked with respect to crane inspection, testing, and maintenance. Where any exception is taken to this standard, sufficient information should be provided to demonstrate the equivalency of proposed alternatives.

   f. Verification that crane design complies with the guidelines of CMAA Specification 70 and Chapter 2-1 of ANSI B30.2-1976, including the demonstration of equivalency of actual design requirements for instances where specific compliance with these standards is not provided.

g. Exceptions, if any, taken to ANSI B30.2-1976 with respect to operator training, qualification, and conduct.

2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE VICINITY OF FUEL STORAGE POOLS

NUREG 0612, Section 5.1.2, provides guidelines concerning the design and operation of load-handling systems in the vicinity of stored, spent fuel. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that in this area, either the likelihood of a load drop which might damage spent fuel is extremely small, or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I through III.

1. Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying loads which could, if dropped, land or fall into the spent fuel pool.

2. Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads or are permanently prevented from movement of the hook centerline closer than 15 feet to the pool boundary, or by providing a suitable analysis demonstrating that for any failure mode, no heavy load can fall into the fuel-storage pool.

3. Identify any cranes listed in 2.2-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6 or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

4. For cranes identified in 2.2-1, above, not categorized according to 2.2-3, demonstrate that the criteria of NUREG 0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the spent fuel area and your determination of compliance. This response should include the following information for each crane:

   a. Which alternatives (e.g., 2, 3, or 4) from those identified in NUREG 0612, Section 5.1.2, have been selected.

b. If Alternative 2 or 3 is selected, discuss the crane motion limitation imposed by electrical interlocks or mechanical stops and indicate the circumstances, if any, under which these protective devices may be bypassed or removed. Discuss any administrative procedures invoked to ensure proper authorization of bypass or removal, and provide any related or proposed technical specification (operational and surveillance) provided to ensure the operability of such electrical interlocks or mechanical stops.

c. Where reliance is placed on crane operational limitations with respect to the time of the storage of certain quantities of spent fuel at specific post-irradiation decay times, provide present and/or proposed technical specifications and discuss administrative or physical controls provided to ensure that these assumptions remain valid.

d. Where reliance is placed on the physical location of specific fuel modules at certain post-irradiation decay times, provide present and/or proposed technical specifications and discuss administrative or physical controls provided to ensure that these assumptions remain valid.

e. Analyses performed to demonstrate compliance with Criteria I through III should conform to the guidelines of Attachment 5. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

2.3 SPECIFIC REQUIREMENTS OF OVERHEAD HANDLING SYSTEMS OPERATING IN THE CONTAINMENT

NUREG 0612, Section 5.1.3, provides guidelines concerning the design and operation of load-handling systems in the vicinity of the reactor core. Information provided in response to this section should be sufficent to demonstrate that adequate measures have been taken to ensure that in this area, either the likelihood of a load drop which might damage spent fuel is extremely small, or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I through III.

1. Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., taking no credit for any interlocks or operating procedures) of carrying heavy loads over the reactor vessel.

2. Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads, or are permanently prevented from the movement of any load either directly over the reactor vessel or to such a location where in the event of any load-handling-system failure, the load may land in or on the reactor vessel.

3. Identify any cranes listed in 2.3-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

4. For cranes identified in 2.3-1, above, not categorized according to 2.3-3, demonstrate that the evaluation criteria of NUREG 0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in your response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the containment and your determination of compliance. This response should include the following information for each crane:

   a. Where reliance is placed on the installation and use of electrical interlocks or mechanical stops, indicate the circumstances under which these protective devices can be removed or bypassed and the adminstrative procedures invoked to ensure proper authorization of such action. Discuss any related or proposed technical specification concerning the bypassing of such interlocks.

   b. Where reliance is placed on other, site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications and discuss administrative or physical controls provided to ensure the continued validity of such considerations.

   c. Analyses performed to demonstrate compliance with Criteria I through III should conform with the guidelines of Attachment 5. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

2.4 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, CORE DECAY HEAT REMOVAL, OR SPENT FUEL POOL COOLING

NUREG 0612, Section 5.1.5, provides guidelines concerning the design and operation of load-handling systems in the vicinity of equipment or components

required for safe reactor shutdown and decay heat removal. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that in these areas, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or that damage to such equipment from load drops will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1, and their loads in your response to 2.1-3-c.

1. Identify any cranes listed in 2.1-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial comliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

2. For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.4-1, a comprehensive hazard evaluation should be provided which includes the following information:

   a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1-3-c. Impact areas should be identified by construction zones and elevations or by some other method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

   b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned considerations should be supplemented by the following specific information:

      (1) For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

(2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the bypassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

(3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance sequencing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the continued validity of such considerations.

c. For interactions not eliminated by the analysis of 2.4-2-b, above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

d. For interactions not eliminated in 2.4-2-b or 2.4-2-c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG 0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided.

(1) An indication of whether or not, for the specific load being investigated, the overhead crane-handling system is designed and constructed such that the hoisting system will retain its load in the event of seismic accelerations equivalent to those of a safe shutdown earthquake (SSE).

(2) The basis for any exceptions taken to the analytical guidelines of Attachment 5.

(3) The information requested in Attachment 4.

FIGURE 1

Typical Load/Impact Area Matrix

CRANE: (IDENTIFY THE CRANE BY NAME AND EQUIPMENT NUMBER)

LOCATION	INDICATE THE BUILDING(S) CORRESPONDING TO THE IMPACT AREA(S) EXAMPLE: REACTOR BUILDING, AUXILIARY BUILDING					
IMPACT AREA / LOADS	(IDENTIFY AREA BY CONSTRUCTION ZONES) Example: Column Line F-S, Column Line R9-R12					
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
(Heavy Load Identification should include designation and weight) Example Spent Fuel Cask WL: 10/24 (100 tons)	(Indicate the various elevations) Example: Elev. 435'	Note 1	Note 2			

NOTES TO FIGURE 1

Note 1: Indicate by symbols the safety-related equipment. The licensee should provide a list consistent with the clarification provided in 1.2-3.

Note 2: <u>Hazard Elimination Categories</u>

a. Crane travel for this area/load combination prohibited by electrical interlocks or mechanical stops.

b. System redundancy and separation precludes loss of capability of system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations eliminate the need to consider load/equipment combination.

d. Likelihood of handling system failure for this load is extremely small (i.e. section 5.1.6 NUREG 0612 satisfied).

e. Analysis demonstrates that crane failure and load drop will not damage safety-related equipment.

SINGLE-FAILURE-PROOF HANDLING SYSTEMS

1. Provide the name of the manufacturer and the design-rated load (DRL). If the maximum critical load (MCL), as defined in NUREG 0554, is not the same as the DRL, provide this capacity.

2. Provide a detailed evaluation of the overhead handling system with respect to the features of design, fabrication, inspection, testing, and operation as delineated in NUREG 0554 and supplemented by the identified alternatives specified in NUREG 0612, Appendix C. This evaluation must include a point-by-point comparison for each section of NUREG 0554. If the alternatives of NUREG 0612, Appendix C, are used for certain applications in lieu of complying with the recommendation of NUREG 0554, this should be explicitly stated. If an alternative to any of those contained in NUREG 0554 or NUREG 0612, Appendix C, is proposed details must be provided on the proposed alternative to demonstrate its equivalency.

3. With respect to the seismic analysis employed to demonstrate that the overhead handling system can retain the load during a seismic event equal to a safe shutdown earthquake, provide a description of the method of analysis, the assumptions used, and the mathematical model evaluated in the analysis. The description of assumptions should include the basis for selection of trolley and load position.

4. Provide an evaluation of the lifting devices for each single-failure-proof handling system with respect to the guidelines of NUREG 0612, Section 5.1.6.

5. Provide an evaluation of the interfacing lift points with respect to the guidelines of NUREG 0612, Section 5.1.6.

ANALYSIS OF RADIOLOGICAL RELEASES

The following information should be provided for an analysis conducted to demonstrate compliance with Criterion I of NUREG 0612, Section 5.1.

1. INITIAL CONDITIONS/ASSUMPTIONS

   a. Identify the time after shutdown, the number of fuel assemblies damaged, and the assumed duration of radiological release associated with each accident analyzed.

   b. NUREG 0612, Table 2.1-2, provides the assumptions used to arrive at generic conclusions concerning radiological dose consequences. To rely on the radiological dose analysis of NUREG 0612, the licensee should verify that these assumptions are conservative with regard to the plant/site evaluated. If the assumptions are not conservative for the specific plant, or if a more site-specific analysis is required, the licensee should identify plant-specific assumptions used in place of those tabulated.

   c. Identify and provide the basis (e.g., USNRC Regulatory Guide 1.25) for any assumptions employed in site-specific analyses not identified in NUREG 0612, Table 2.1-2.

   d. Dose calculations based on the termination or mitigation of radiological releases should be supported by information sufficient to demonstrate both that the time delay assumed is conservative and that the system provided to accomplish such termination or mitigation will perform its safety function upon demand (i.e., the system meets the criteria for an Engineered Safety Feature). Specific information so provided should include the following:

      (1) Details concerning the location of accident sensors, parameters monitored and the values of these parameters at which a safety signal will be initiated, system response time (including valve-operation time), and the total time required to automatically shift from normal operation to isolation or filtration following an accident.

      (2) A description of the instrumentation and controls associated with the Engineered Safety Feature which includes information sufficient to demonstrate that the requirements (Section 4) of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," are satisfied.

(3) A description of any Engineered Safety Feature filter system which includes information sufficient to demonstrate compliance with the guidelines of USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

(4) A discussion of any initial conditions (e.g., manual valves locked shut, containment airlocks or equipment hatches shut) necessary to ensure that releases will be terminated or mitigated upon Engineered Safety Feature actuation and the measures employed (i.e., Technical Specification and administrative controls) to ensure that these initial conditions are satisfied and that Engineered Safety Feature systems are operable prior to the load lift.

2. METHOD OF ANALYSIS

Discuss the method of analysis used to demonstrate that post-accident dose will be well within 10CFR100 limits. In presenting methodology used in determining the radiological consequences, the following information should be provided.

a. A description of the mathematical or physical model employed.

b. An identification and summary of any computer program used in this analysis.

c. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

3. CONCLUSION

Provide an evaluation comparing the results of the analysis to Criterion I of NUREG 0612, Section 5.1. If the postulated heavy-load-drop accident analyzed bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

CRITICALITY ANALYSIS

The following information should be provided for analyses conducted to demonstrate compliance with Criterion II of NUREG 0612, Section 5.1

1. INITIAL CONDITIONS/ASSUMPTIONS

The conclusions of NUREG 0612, Section 2.2, are based on a particular model fuel assembly. If a licensee uses the results of Section 2.2 rather than performing an independent neutronics analysis, the assumptions should be verified to be compatible with plant-specific design. For any analysis conducted, the following assumptions should be provided as a minimum:

a. Water/UO_2 volume ratio

b. The boron concentration for the refueling water and spent-fuel pool

c. The amount of neutron poison in the fuel

d. Fuel enrichment

e. The reactivity insertion value due to crushing of the core

f. The k_{eff} value allowed by technical specifications for the core during refueling

2. METHOD OF ANALYSIS

Provide the method of analysis used to demonstrate that accidental dropping of a heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95. The discussion of the method of analysis should include the following information:

a. Identification of the computer codes employed

b. A discussion of allowances or compensation for calculation and physical uncertainties

3. CONCLUSION

Provide an evaluation comparing the results of the analysis to Criterion II of NUREG 0612, Section 5.1. If the postulated heavy-load-drop accident

bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

Attachment (4)

ANALYSIS OF PLANT STRUCTURES

The following information should be provided for analyses conducted to demonstrate compliance with Criteria III and IV of NUREG 0612, Section 5.1.

1. INITIAL CONDITIONS/ASSUMPTIONS

Discuss the assumptions used in the analysis, including:

a. Weight of heavy load

b. Impact area of load

c. Drop height

d. Drop location

e. Assumptions regarding credit taken in the analysis for the action of impact limiters

f. Thickness of walls or floor slabs impacted

g. Assumptions regarding drag forces caused by the environment

h. Load combinations considered

i. Material properties of steel and concrete

2. METHOD OF ANALYSIS

Provide the method of analysis used to demonstrate that sufficient load-carrying capability exists within the wall(s) or floor slab(s). Identify any computer codes employed, and provide a description of their capabilities. If test data was employed, provide it and describe its applicability.

3. CONCLUSION

Provide an evaluation comparing the results of this analysis with Criteria III and IV of NUREG 0612, Section 5.1. Where safe-shutdown equipment has a ceiling or wall separating it from an overhead handling system, provide an evaluation to demonstrate that postulated load drops do not penetrate the ceiling or cause secondary missiles that could prevent a safe-shutdown system from performing its safety function.

APPENDIX A

ANALYSES OF POSTULATED LOAD DROPS

Certain of the alternatives in Sections 5.1.2 through 5.1.5 of this report call for an analysis of postulated load drops and evaluation of potential consequences to assure that the evaluation criteria of Section 5.1 are met for such an event. Section A-1 of this appendix identifies certain considerations that should be included in such evaluations. Sections A-2 and A-3 identify certain additional considerations and assumptions that should be used in analyzing the potential consequences of a drop of the reactor vessel head assembly or the spent fuel shipping cask; other load drops that are analyzed should use similar considerations and assumptions that are appropriate for these other loads. Section A-4 provides guidance in performing criticality calculations.

1. GENERAL CONSIDERATIONS

Analyses of postulated load drops should as a minimum include the considerations listed below. Other considerations may be appropriate for the particular load drop being analyzed; for example, for a reactor vessel head assembly or a spent fuel cask drop analysis, the additional considerations listed in Sections A-2 or A-3 should be used. In evaluating the potential for a load drop to result in criticality, the considerations of A-4 should also be followed. The following should be considered for any load drop analysis, as appropriate:

(1) That the load is dropped in an orientation that causes the most severe consequences;

(2) That fuel impacted is 100 hours subcritical (or whatever the minimum that is allowed in facility technical specifications prior to fuel handling);

(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;

(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;

(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.

(6) Analyses should be based on an elastic-plastic curve that represents 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

(8) Loads need not be analyzed if their load paths and consequences are scoped by the analysis of some other load.

(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.

(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.

2. REACTOR VESSEL HEAD DROP ANALYSIS*

Where a reactor vessel head drop analysis is to be performed to satisfy the PWR Containment or BWR Reactor Building guidelines (Sections 5.1.3 or 5.1.4) of this report, the analysis should consider the following to assure that the evaluation criteria of Section 5.1 are satisfied.

(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).

(2) All potential accident cases during the refueling operations. 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.

(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.

*These guidelines only consider the dropping of the RV head assembly during refueling and do not apply directly to dropping of the reacter 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.

(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.

(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.

(6) The RV head assembly should be considered rigid and not experience deformation during impact with other components or structures.

3. SPENT FUEL CASK DROP ANALYSIS

Where a cask drop analysis is to be performed to satisfy the guidelines in Sections 5.1.2, 5.1.4, or 5.1.5 of this report, it should consider the following in addition to the general considerations of Section A-1 to assure that the evaluation criteria of Section 5.1 are satisfied:

(1) Applying a single-failure to the lifting assembly, consider that the cask is dropped in an orientation that will result in the most severe consequences.

(2) Impact loads should include a fully loaded cask (with water, where applicable) and all equipment required for lifting and set down such as baseplates, lifting yokes, wire ropes and crane blocks.

(3) Restricted path travel of the spent fuel cask (defined by electrical interlocks, mechanical stops, and crane travel capability) should be evaluated to determine the locations and probable accident cases along the path where damage could occur to:

(a) the floor and walls of the Spent Fuel Pool (SFP);
(b) racks within the SFP which support the spent fuel;
(c) the spent fuel itself;
(d) the refueling channel gate; or
(e) safety related systems, components and structures beneath or adjacent to the travel path of the cask.

(4) In the analysis consideration may be given to drag forces caused by the environment of the postulated accident case, e.g., when the spent fuel cask is postulated to drop into the SFP, credit may be taken for drag forces caused by the water in the SFP. Water level assumed for such analyses should be the minimum level allowed by technical specifications.

(5) Credit may be taken for energy absorbing devices integral to the cask if attached during the handling operations in determining the amount of energy imparted to the spent fuel or safety related systems, components or structures.

(6) For the purpose of the analysis the cask should be considered rigid (except for devices and appurtences specifically designed for energy absorption and in place) and not to experience deformation during impact.

(7) In the calculating the center of gravity, consideration should be given to modifications made to the cask after purchase, e.g., addition of a perforated metal basket within the cask.

4. CRITICALITY CONSIDERATIONS

4.1 Spent Fuel Pool Neutronics Analysis

In Sections 5.1.2, "Spent Fuel Pool Area - PWR," and 5.1.4, "Reactor Building - BWR," a number of alternatives are presented for the control of heavy loads in spent fuel pool areas. Some of these alternatives include neutronics calculations to demonstrate that crushing the fuel and fuel rack will not result in criticality. This section is included here to give the licensees guidance in performing their neutronics calculation.

A discussion of the potential for criticality under load drop conditions is discussed in Section 2.2, and summarized in Section 2.2.6. The results of this section should be used as a guide to determine which neutronics or other analyses are required to evaluate the potential for criticality for a specific plant area. A licensee may choose to use the results of section 2.2, rather than performing an independent neutronics analysis for his plant. If a licensee uses the results of Section 2.2 rather than performing an independent neutronics analysis, he should verify that the assumptions and model fuel assembly of Section 2.2 are valid for his plant.

For PWR spent fuel pools, credit may be taken under the accident conditions of a load drop for the boron in the spent fuel pool water to maintain subcriticality. In this case the required boron concentration should be specified in the facility Technical Specification, and regular monitoring of the boron concentration in the spent fuel pool should also be specified. Likewise, if the neutronics analysis postulates a bounding distribution of non-spent fuel within the spent fuel pool, then the Technical Specifications must be modified to require that the actual distribution of fuel is no more deleterious than that assumed in the analysis. In postulating a limiting distribution of non-spent fuel, the licensee may either assume an infinite array or a finite array. The largest finite array of non-spent fuel a licensee should have to consider would be that of an off-load core.

In this neutronics analysis the licensee must demonstrate that the fuel remains subcritical in the optimum crushed configuration. It is adequate to assume that the optimum configuration is with the rack crushed to uniformly reduce the separation between assemblies and the spacing between fuel pins uniformly reduced to maximize k_{eff}. All boral and structural material may be assumed to remain in its original configuration relative to the fuel, and not forced out of the fuel array.

The neutronics analysis for the spent fuel pool should consider the case where it has become necessary to off-load an entire core into the spent fuel pool and a heavy load is dropped on fuel in the pool.

As noted in Section 5.1.4 it is not necessary to analyze the effects of crushing on k_{eff} for BWR spent fuel pools that use boron plate cans and do not rely on spacing to maintain subcriticality.

4.2 Reactor Core Neutronics Analyses

4.2.1 Neutronics Analyses for a BWR Core

For a BWR core, the potential for a load drop to drive control rods out of the core should be analyzed using the appropriate considerations of Sections A-1 and A-2. If this analysis shows that postulated load drops could drive control rods out of the core, the number of rods that could be affected should be determined, and a neutronics analysis performed to determine the potential for criticality to result. If in the analysis it is assumed that all rods are in the core just prior to the load drop, then the facility technical specifications should require that all rods are in when handling a heavy load over the core.

4.2.2 Neutronics Analyses for a PWR Core

In Table 2.2-2, we see that crushing the model PWR core in 2000 ppm boron refueling water increases k_{eff} by about 0.02. Since only one model fuel geometry was considered here, other fuel geometries could have a slightly higher reactivity insertion due to crushing. A value of 0.05 may be used as a bounding worst case reactivity insertion value due to crushing of a PWR core. In performing a neutronics evaluation of a postulated load drop on a PWR core, a licensee may use this estimated reactivity insertion limit in lieu of performing a plant specific calculation. If a licensee can demonstrate that for his fuel a value less than 0.05 is bounding, then he may use this lower value instead.

The current Technical Specifications require that during refueling k_{eff} should be maintained at 0.95 or less. This is based on an uncrushed core. To perform a neutronics analysis to demonstrate that crushing the core will not drive it critical at least two alternatives for demonstrating this are acceptable.

(1) The licensee can perform a neutronics analysis on his core uniformly crushed in the x-y direction to maximize k_{eff}. If the licensee chooses this option he must demonstrate that the maximum k_{eff} is no greater than 0.95, with all uncertainties taken into account.

OR

(2) Using his core refueling neutronics analysis (uncrushed), the licensee can demonstrate that k_{eff} for the uncrushed core is no greater than 0.90. Then, using the estimated 0.05 maximum reactivity insertion due to crushing, the maximum achievable k_{eff} is still less than 0.95.

5. ACCEPTANCE CRITERIA

In performing the above analyses, the acceptance criteria for resultant damage should be that it does not cause a condition that may exceed evaluation criteria I-IV stated in Section 5.1 of this report.

TABLE 3.1-1

SURVEY OF HEAVY LOADS

Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
1. PWR - Refueling Building	1.	Spent Fuel Shipping Cask	(P)	15-110 Tons (13-100,000 kg)	2/
	2.	Pool Divider Gates (some plants)	(P)	2 Tons (1800 kg)	2-4 x's (per refueling)
	3.	Fuel Transfer Canal Door	(P)	2 Tons (1800 kg)	2-4 x's (per refueling)
	4.	Missile Shields	(P)	4-20 Tons (4-19,000 kg)	2 x's (per refueling)
	5.	Irradiated Specimen Shipping Cask	(P)	3.5-12 Tons (3-11,000 kg)	Once per year to once per 10 years
	6.	Plant Equipment (some plants) (e.g., pumps, motors, valves, heat exchangers, etc.)	(O)	2-4 Tons (1800-3600 kg)	As required for modification or replacement
	7.	Spent resin, filter, or other radioactive material shipping casks	(P)	5-37 Tons (4500-33,000 kg)	~ 5 x's per year
	8.	New fuel shipping containers with fuel (usually 4 assemblies)	(P)	3-4 Tons (2700-3300 kg)	3/
	9.	Failed Fuel Container	(O)	1 Ton (900 kg)	Less than once per refueling

TABLE 3.1-1 (Continued)

Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
1. (cont.)	10.	Fuel transfer carriage	(O) or (P)	1.5 Tons (1300 kg)	Only for maintenance or repair (~ once per 10 years)
	11.	Crane Load Block	(O)	4-10 Tons (4-9,000 kg)	1/
2. PWR - Containment Building	1.	Reactor Vessel Head	(O)	55-165 Tons (50-150,000 kg)	2 x's (per refueling)
	2.	Upper Internals	(O)	25-65 Tons (23-33,000 kg)	2 x's (per refueling)
	3.	In-Service Inspection Tool	(O)	4.5 Tons (4,000 kg)	Used at least once every three years
	4.	Reactor Coolant Pump	(P)	30-40 Tons (27-36,000 kg)	4-10 x's over life of plant
	5.	Missile Shields	(P)	10-20 Tons (9-18,000 kg)	2 x's (per refueling)
	6.	Crane Load Block	(O)	4-10 Tons (4-9,000 kg)	1/
3. BWR-- Reactor Building	1.	Missile or Shield Plugs (6-12)	(P)	15-125 Tons (13-112,000 kg)	2 x's (per refueling)
	2.	Drywell Head	(P)	45-85 Tons (40-77,000 kg)	2 x's (per refueling)

TABLE 3.1-1 (Continued)

Area	Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
3. (cont.)	3. Reactor Vessel Head	(O) (Over reactor)	45-96 Tons (40-86,000 kg)	2 x's (per refueling)
	4. Steam Dryers 5/	(O) (Over reactor)	20-40 Tons (18-36,000 kg)	2 x's (per refueling)
	5. Moisture Separators 5/	(O) (Over reactor)	20-75 Tons (18-68,000 kg)	2 x's (per refueling)
	6. Spent Fuel Pool Gates	(O) (Over spent fuel pool)	2-6 Tons (1800-5,000 kg)	2 x's (per refueling)
	7. Dryer/Separator Storage Pit Shield Plugs (some plants)	(P)	75 Tons (68,000 kg)	2 x's (per refueling)
	8. Refueling Slot Plugs	(O) (Over spent fuel pool)	2-6 Tons (1800-5400 kg)	2 x's (per refueling)
	9. Spent Fuel Shipping Cask	(O) (Over spent fuel pool)	15-110 Tons (14-99,000 kg)	4/
	10. Vessel Service Platform	(O)	1-5 Tons (900-4500 kg)	5-10 x's (per refueling)
	11. Waste and Debris Shipping Casks	(O) (Over reactor and/ or spent fuel pool)	8-30 Tons (7-27,000 kg)	1-3 x's (per year)

TABLE 3.1-1 (Continued)

Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
3. (cont.)	12.	Vessel Head Insulation	(P)	4-6 Tons (4-5,000 kg)	2 x's (per refueling)
	13.	Replacement Fuel Storage Racks for Spent Fuel	(O) (Over spent fuel)	8 Tons (7,000 kg)	On installation
	14.	Crane Load Block	(O)	4-10 Tons (4-9,000 kg)	1/
	15.	Plant Equipment	(O) (Over safety equip.)	1 Ton (900 kg)	
4. Other Plant Areas	1.	Spent Fuel Shipping Casks (some plants)	(O) (Over safety equipment)	15-110 Tons (14-99,000 kg)	2/, 4/
	2.	Turbine or other equipment in turbine building (some plants)	(O) (Over) safety equipment)	2-150 Tons (2-135,000 kg)	As required for equipment overhaul and replacement
	3.	Other plant equipment (pumps, motors, valves, heat exchangers, etc.)	(O) (Over safety equipment)	1-30 Tons (1-27,000 kg)	As required for equipment overhaul and replacement

TABLE 3.1-1

FOOTNOTES

1/ Listed weight for loads does not include weight of load block except where listed separately. The load block may add 4-10 tons (4,000 - 9,000 kg) to the weight of the dropped load. Because of this, the load block should be considered a heavy load even if it is not carrying a load, or is being used with a lighter load.

2/ These are presently not being used at most plants. However, once offsite waste repositories are established, casks will be used frequently for shipping spent fuel offsite. For a typical 1,000 MWe pressurized water reactor, spent fuel casks must be shipped offsite from 7 to 65 times per year depending on the size cask used. This is based on casks currently licensed for use in the United States.

3/ A typical 1,000 MWe power plant would usually require 16 or 17 new fuel containers (four fuel assemblies each) per year.

4/ These are presently not being used at most plants. However, once offsite waste repositories are established, casks will be used frequently for shipping spent fuel offsite. For a typical 1,000 MWe boiling water reactor, spent fuel casks must be shipped offsite from 12 to 125 times per year depending on the size cask used. This is based on casks currently licensed for use in the United States.

5/ Due to certain dimensional restrictions, for most BWR's it would not be possible to drop the dryers or moisture separators onto fuel in the reactor core.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

I - Fuel (New and Spent)

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
4986	RA-1, 2, 3, J	General Electric Co.		TVA
5450	RCC, 1, 2, 3	Westinghouse Electric		VEP, DLC
5805	Vandenburgh	Chem-Nuclear Systems, Inc.	70,000	APC, CPL, DLP, DPC, FPL, FPC, JCP, NPP, VEP
5901	NFS Model 100	Nuclear Fuel Services	126,200	CPC, PGE
5938	HNPF		48,000	PEC
6078	927A1 927C1	Combustion Engineering, Co.	6200 7000	APL
6206	B	Babcock & Wilcox Co.	6940	DPC, FPC
6273	48 (Series)		4500	VEP
6375	PB-1	Chem-Nuclear Systems, Inc.	67,050	APC, BEC, CPL, DPC FPL, FPC, GPC, JCP, MYA, MEC, NNE, NSP, PNY, TVA, VEP
6400	Super Tiger	Westinghouse Electric Co.	45,000	APL, CPC, DLP, DLC, MEC, NPP, SMU, VEP
6698	NFS-4	Nuclear Fuel Services, Inc.	50,000	BGE, BEC, CWE, DLP, DPC, FPL, FPC, JCP, MYA, RGE, SCE, WMP,
9001	IF 300	General Electic Co.	140,000	CPL, CWE
9010	NLI-1/2	NL Industries, Inc.	47,500	BEC, FPL, VYC
9044	GE-1600	General Electric Co.	23,000	APC, BGE, BEC, CPL, CPC, DPC, FPL, FPC, GPC, IEL, JCP, MEC, NNE, NSP, VEP, VYC

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
5026	BC-48-220	Chem-Nuclear Systems, Inc.	71,000	APC, BEC, CPL, CWE, CYA, DPC, DLC, FPL, FPC, JCP, NPP, VEP, WPS
6058	B3-1	Nuclear Engineering Co.	30,000	APL, CPC, DLP, IEL, MEC, NPP, NSP, PGE, SMU, TEC, VEP
6144	6144	Nuclear Engineering Co.	42,000	APC, APL, CPL, CEC, CPC, DLP, DPC, FPL, FPC, GPC, IEL, JCP, MEC, NPP, NSP, PGE, PNY, RGE, SMU, VEP
6244	6244	Chem-Nuclear Systems, Inc.	46,000	APC, CPL, CWE, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NSP, PEC, VEP, WMP
6272	Poly Panther	Nuclear Engineering Co.	6100	APL, CPC, DLP, MEC NPP, SMU, VEP
6568	LL-60-150	Tennessee Valley Auth.	73,000	
6574	HN 200	Hittman Nuclear and Development Corp.	47,000	APL, BGE, CWE, CEC, DLP, DLC, IME, JCP, MYA, MEC, NPP, PEC, PNY, VYC, YAC
6601	LL-50-100	Chem-Nuclear Systems, Inc.	70,000	APC, BEC, CPL, CYA, CEC, CPC, DLP, DPC, FPL, FPC, JCP, NPP, NNE, PEG, RGE, TVA, VEP
6679	1/2 Super Tiger	Nuclear Engineering Co.	45,000	APL, CPC, DLP, MEC, NPP, SMU, VEP
6722	BS-33-180	Tennessee Valley Auth.	51,000	

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
6744	Poly Tiger	Nuclear Engineering Co.	35,000	APL, BEC, CPC, DLP, MEC, NPP, SMU, VEP
6771	SN-1	Nuclear Engineering Co.	60,000	APL, CPC, DLP, NPP, SMU, VEP
9074	AP-100		28,000	DLC
9079	HN-100 Ser. 2	Hittman Nuclear and Development Corp.	98,000	APL, BGE, CEC, CWE, DLP, IME, JCP, MYA, MEC, NPP, PEC
9080	HN-600	Hittman Nuclear and Development Corp.	42,000	BGE, CWE, CEC, DLP, IME, IEL, JCP, MYA, MEC, NPP, PEC, YAC
9086	HN-100 Ser. 1	Hittman Nuclear and Development Corp.	46,000	APL, BGE, CWE, DLP, IME, JCP, MYA, MEC, NPP, NNE, PEC, RGE, VYC
9089	HN-100S	Hittman Nuclear and Development Corp.	36,500	BGE, CWE, CEC, IME, JCP, MYA, NPP, PEC
9092	HN-300	Hittman Nuclear and Development Corp.	43,000	MYA
9093	HN-400	Hittman Nuclear and Development Corp.	43,000	MYA
9094	CNSI-14-195-H	Chem-Nuclear Systems, Inc.	56,500	APC, APL, BEC, CPL, CWE, CYA, CEC, CPC, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, NSP, OPP, PGE, PEC, PGC, PNY, PEG, TVA, VEP
9096	CNSI-21-300	Chem-Nuclear Systems, Inc.	57,450	APC, APL, CPL, CEC, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, PNY, PEG, VEP

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
9105	RAD-Waste CR.I	Chem-Nuclear Systems, Inc.	58,400	APC, CPL, DPC, FPL, FPC, GPC, JCP, MEC, NMP, VEP
9108	AL-33-90	Chem-Nuclear Systems, Inc.	41,300	APC, CPL, CWE, CEC, DPC, FPL, FPC, JCP, NPP, NMP, NNE, PGC, VEP, WEP
9111	CN6-80A	Chem-Nuclear Systems, Inc.	51,500	APC, CPL, CWE, CEC, DPC, FPL, FPC, GPC, MEC, NNE, PGC, SMU, VEP
9113	7-100	Chem-Nuclear Systems, Inc.	7000	APC, BEC, CPL, CWE, CYA, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, NSP, VEP
9122	18-450	Chem-Nuclear Systems, Inc.	61,000	BEC

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

III - Byproducts

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
5971	GE-200		10,000	PEC
5980	GE-600		18,500	NNE, NSP
6275	LL-28-4	Chem-Nuclear Systems, Inc.	30,000	APC, CPL, DPC, FPL, FPC, NPP, VEP
9081	CNS-1600	Chem-Nuclear Systems, Inc.	26,000	APC, BGE, CPL, DPC, FPL, FPC, GPC, NSP, TVA, VEP

*See attached list of abbreviations.

LICENSEE ABBREVIATIONS

APC	Alabama Power Company
APL	Arkansas Power and Light Company
BEC	Boston Edison Company
BGE	Baltimore Gas and Electric Company
CEC	Consolidated Edison Company
CPC	Consumers Power Company
CPL	Carolina Power and Light Company
CWE	Commonwealth Edison Company
CYA	Connecticut Yankee Atomic Power Company
DLC	Duquesne Light Company
DLP	Dairyland Power Cooperative
DPC	Duke Power Company
FPC	Florida Power Corporation
FPL	Florida Power and Light Company
GPC	Georgia Power Company
IEL	Iowa Electric Light and Power Company
IME	Indiana and Michigan Electric Company
JCP	Jersey Central Power and Light Company
MEC	Metropolitan Edison Company
MYA	Maine Yankee Atomic Power Company
NMP	Niagara Mohawk Power Corporation
NNE	Northeast Nuclear Energy Company
NPP	Nebraska Public Power Corporation
NSP	Northern States Power Company
OPP	Omaha Public Power District
PEC	Philadelphia Electric Company
PEG	Public Service Electric and Gas Company
PGC	Portland General Electric Company
PNY	Power Authority of the State of New York
RGE	Rochester Gas and Electric Corporation
SMU	Sacramento Municipal Utilities Corporation
TEC	Toledo Edison Company
TVA	Tennessee Valley Authority
VEP	Virginia Electric and Power Company
VYC	Vermont Yankee Nuclear Power Corporation
YAC	Yankee Atomic Electric Company
WMP	Wisconsin-Michigan Power Company
WPS	Wisconsin Public Service Corporation