7/31/80 Docket file 50-286

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Docket No. [redacted]

REGULATORY DOCKET FILE COPY

Mr. George T. Berry
President and Chief Operating
Officer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

In order to complete our review of the analysis of the reactor coolant system for postulated loss of coolant accident (LOCA) for Indian Point No. 3, we require that you assess the effects of combining the seimsic (SSE) and LOCA responses. We have reviewed the currently available information to support decoupling of these two events and have determined that currently a sufficient basis for decoupling these events does not exist in the nuclear industry.

In cases where the SSE responses have been calculated elastically and the LOCA responses have been calculated inelastically, an acceptable method of computing the combined responses is to combine the LOCA and SSE strain components absolutely.

Therefore, within 30 days of the date of this letter, provide your assessment of the effects of this load combination on the analysis presented in WCAP-9117, including the adequacy of the proposed pipe whip restraints design. In the event the design is found to be adequate for such loads, we recommend that you proceed with the installation of the restraints during your next refueling outage.

Sincerely,

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

8008150 288

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Docket No. 50-286

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President and Chief Operating
Officer
Power Authority of the State
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10 Columbus Circle
New York, New York 10019

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In cases where the SSE responses have been calculated elastically and the LOCA responses have been calculated inelastically, an acceptable method of computing the combined responses is to combine the LOCA and SSE strain components absolutely.

Therefore, within 30 days of the date of this letter, provide your assessment of the effects of this load combination on the analysis presented in WCAP-9117, including the adequacy of the proposed pipe whip restraints design. In the event the design is found to be adequate for such loads, we recommend that you proceed with the installation of the of the restraints during your next refueling outage.

Sincerely,

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

*SEE PREVIOUS CONCURRENCE

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DATE	7/ /80	7/[illegible]/80	7/30/80	7/30/80	7/31/80	

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Docket No. 50-286

Mr. George T. Berry
President and Chief Operating
  Officer
Power Authority of the State
  of New York
10 Columbus Circle
New York, New York 10019

responses

Dear Mr. Berry:

In order to complete our review of the analysis of the reactor coolant system for postulated loss of coolant accident (LOCA) for Indian Point, No. 3 we require that you assess the effects of combining the seismic (SSE) and ~~LOCA events~~. We have reviewed the currently available information to support decoupling of these two events and have determined that currently a sufficient basis for decoupling these events does not exist in the nuclear industry.

responses

of combining the
of computing combined responses

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Therefore, within 30 days of the date of this letter, ~~justify that the pipe whip restraints proposed in WCAP-9117 are adequately designed to accommodate the LOCA plus SSE loads~~. In the event the design is found to be adequate for such loads, we recommend that you proceed with the installation of the restraints during your next refueling outage.

provide your assessment of the effects of this load combination on the analysis presented in WCAP-9117, including the adequacy of the proposed pipe whip restraints design.

Sincerely,

Thomas M. Novak, Assistant Director
  for Operating Reactors
Division of Licensing

cc: See next Page

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DATE	7/29/80					

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

Docket No. 50-286

July 31, 1980

Mr. George T. Berry
President and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

In order to complete our review of the analysis of the reactor coolant system for postulated loss of coolant accident (LOCA) for Indian Point No. 3, we require that you assess the effects of combining the seimsic (SSE) and LOCA responses. We have reviewed the currently available information to support decoupling of these two events and have determined that currently a sufficient basis for decoupling these events does not exist in the nuclear industry.

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Sincerely,

T M Novak

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Mr. George T. Berry
Power Authority of the State of New York

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White Plains, New York 10601

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Assistant General Counsel
Power Authority of the
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Sheldon, Harmon and Weiss
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Dr. Lawrence D. Quarles
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Mr. George M. Wilverding
Licensing Supervisor
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Mr. P. W. Lyon, Senior Vice
President - Nuclear Generation
Power Authority of the
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New York, New York 10019

Mr. J. P. Bayne, Resident Manager
Indian Point 3 Nuclear Power Plant
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Mr. J. W. Blake, Ph.D., Director
Environmental Programs
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10 Columbus Circle
New York, New York 10019

Theodore A. Rebelowski
Resident Inspector
Indian Point Nuclear Generating
U. S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

JULY 3 1 1980

Docket No. 50-286

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

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.

W

OFFICE						
SURNAME						
DATE						

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:

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

7/22/80

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As stated

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DATE	07/08/80	07/08/80	07/ /80	07/ /80		

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

July 31, 1980

Docket No. 50-286

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

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

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Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
As stated

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10 Columbus Circle
New York, New York 10019

Theodore A. Rebelowski
Resident Inspector
Indian Point Nuclear Generating
U. S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

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 loads, 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.)

---

* 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 (2) 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 5.1 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) <u>Special lifting devices</u> 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) <u>Lifting devices that are not specially designed</u> 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;

<u>OR</u>

(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) <u>New cranes</u> 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) <u>Interfacing lift points</u> 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.

<u>OR</u>

(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.

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 effort (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 B30.9-1971 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 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.

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 P-S, Columnline 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 NLI 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.

7/31/80

Meeting 6

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