NUREG-1774

A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

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Manuscript Completed: June 2003
Date Published: July 2003

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ABSTRACT

This report was written in response to a candidate generic issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," to determine the likelihood and significance of heavy load drops. This report describes the results of a detailed review of crane operating experience at U.S. nuclear power plants from 1968 through 2002. Crane operating experience information was obtained from several sources including; actual crane operating experience from U.S. nuclear power plants, licensee event reports (10 CFR 50.72 and 10 CFR 50.73), NRC inspection reports, licensee correspondence, and crane vendor reports. This report lists the causes and results of documented crane issues, and estimates the probabilities of selected load drop events. To provide additional insights, included with this report are major crane operating experience reports issued by the New Mexico Environmental Evaluation Group, the Department of Energy, the Department of the Navy, and the California Division of Occupational Safety and Health. The operational experience and human performance insights contained in this report can be used to enhance the control of heavy loads to reduce the likelihood of crane accidents, particularly those that have the potential to release radioactive material. This report also will be used as a technical basis for recommendations that may initiate changes to NRC regulatory requirements, programmatic controls, or evaluations of heavy load movements at U.S. nuclear power plants.

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EXECUTIVE SUMMARY

In nuclear plant operation, maintenance, and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop because of human error or crane failure, they could impact on stored spent fuel, fuel in the core, or on equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In some instances, load drops at specific times, locations, and weights could potentially lead to offsite doses that exceed 10 CFR Part 100 limits.

The U.S. Nuclear Regulatory Commission (NRC) has issued several guidance documents regarding lifting of heavy loads at U.S. nuclear power plants.

In April 1999, a candidate generic issue (GI) was proposed by the Office of Nuclear Reactor Regulation (NRR) of the NRC regarding heavy load drops. NRR requested the Office of Nuclear Regulatory Research (RES) within the NRC to evaluate the issue. NRR was concerned that although licensees may be operating within the regulatory guidelines in Generic Letter (GL) 85-11, “Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants NUREG-0612,” they may not be taking action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In other words, licensees may not be taking adequate measures, if any, to assess and mitigate the consequences of dropped heavy loads.

In May 1999, RES informed NRR that the candidate GI was accepted, and was given the title GI-186, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants.” The candidate GI received an initial screening in accordance with NRC Management Directive 6.4, “Generic Issue Program” by a Reactor Generic Issue Review Panel. This report documents the results of an assessment of the GI, and includes a comprehensive retrieval and analysis of crane operating experience in the U.S. nuclear industry, documented in NRC records and other sources, from 1968 through 2002. Of particular interest was operating experience with very heavy loads (e.g., loads in excess of 27 metric tons [30 tons]) because of the increased likelihood that if dropped from a sufficient height, could also penetrate the floor and disable needed safety equipment. Actual crane operating experience was obtained from nine nuclear facilities (having a total of 19 nuclear power units) representing approximately 13 percent of the available operating experience in the U.S.

The study makes several observations regarding strengths and weaknesses exhibited by crane operating experience and programmatic control of heavy load movements at nuclear power plants. For example:

The human error rate for crane operating events has significantly increased. The percentage of crane issue reports caused by poor human performance has increased over time, and for the last several years, averaged between 70 and 80 percent. Similar human error results were reported in a 1996 DOE report, “Independent Oversight Special Study of Hoisting and Rigging Incidents Within the Department of Energy [DOE].” Human error, whether directly associated with supervisors or equipment operators represented approximately 94 percent of DOE hoisting and rigging incidents. As shown in Navy crane data for 1995-1999, human factors or human errors are the leading causes of Navy crane issues, in that, the categories of improper operation, improper rigging, and procedure failure, accounted for approximately 88 percent of

Navy crane issues. Navy crane equipment failures accounted for approximately 5 percent of crane issues. For U.S. nuclear power plants, the human error rate for very heavy load crane events is less than the human error rate when considering crane events from all load weights (56 percent v.s. 73 percent respectively).

Load drop events have increased in the last decade. During the period 1969-2002, there were 57 reported events involving load drops. Load drops while operating the spent fuel pool crane (representing over half of the load drop events) were largely because of fuel assembly drops caused by grapple operation or human errors which posed no safety issue. Load drops while operating mobile and other cranes (representing almost half of the events) have occurred outside of safety-related areas. However, several load drops have involved overhead cranes similar to those used in safety-related areas of the power plant. When compared to the previous decade (1981-1992), the last decade (1993-2002) experienced a 60 percent increase in the number of load drop events, concurrent with an increase in the number of operating units by 9 percent.

The number of below-the-hook crane events (mainly rigging deficiencies or failures) has greatly increased. For the period 1968 through 2002, there were 47 reported below-the-hook events, many resulting in load drops and damaged equipment. Over the last decade (1993-2002), there were 33 below-the-hook events, of which 17 involved load drops, 10 involved equipment damage, four involved administrative issues, and two involved load slips. During this period, the number of events increased by 230 percent (when compared to the previous decade), concurrent with an increase in the number of operating units by 9 percent.

Calculational methodologies, assumptions, and predicted consequences varied greatly from licensee to licensee for very similar accident scenarios. Accurate load drop analysis is essential, since each licensee uses load drop calculations to determine transport height restrictions which are referenced in their heavy load lift procedures. Load drop analyses also help to determine locations where other measures besides load height restrictions are necessary (e.g., impact limiting devices, interlocks to prevent crane motion over certain areas, or employment of single-failure proof handling systems).

In general, very heavy load drops in BWR plants are more risk significant than very heavy load drops in PWRs because of plant systems layout, in that, for PWRs, spent fuel cask transfer occurs in an area separate from the reactor building and many safety-related systems. However, for BWRs, many very heavy loads are commonly lifted and moved on the upper floor of the reactor building, or the auxiliary building. Should a floor breach occur during a load drop, there are many safety-related components located on lower floors which could be disabled. A load drop in certain areas could simultaneously initiate an accident, and disable accident mitigation equipment. These types of events have the potential to defeat defense-in-depth.

There have been no Accident Sequence Precursor (ASP)events for the period 1985 through 2002 that involved a crane. To be classified as an ASP event, the event must have a conditional core damage probability (CCDP) of at least 1.0E-06. The most risk significant crane events have been those resulting in a loss of power. There have been 10 losses of power events caused by crane operation from 1968 through 2002, nine of which were caused by mobile cranes. Of the nine mobile crane events involving a loss of power, two events had Augmented Inspection Team (AIT) inspections (Palo Verde and Diablo Canyon). During the last decade (1993-2002), there were three events that resulted in a loss of power. This represents a reduction of 43 percent from the preceding decade, concurrent with an increase in the number of operating units by 9 percent.

The number of mobile crane events has declined slightly. There have been 38 recorded events involving mobile crane operation from 1969 through 2002. Many of these resulted in tip overs, load drops, and equipment damage. Several mobile crane events have resulted in a loss or partial loss of power to various electrical lines servicing plant equipment. However, during the last decade (1993 through 2002), an improving mobile crane performance trend occurred with a slight reduction in the number of events when compared to the previous decade (1981 through 1992).

There were three crane events that resulted in radiation exposures. Each was caused by human error. At the Pilgrim facility in 1979, a crane operator lifted an irradiated fuel assembly out of the spent fuel pool, resulting in increased exposure. At Turkey Point Unit 3 in 1992, a maintenance person was inattentive during movement of the polar crane, and fell into the refueling cavity and got contaminated. A third radiation event caused by crane operation occurred at Farley Unit 2 in 1999 when the failure of the polar crane primary height measuring system allowed a portion of the reactor lower internals to be exposed during a lift. None of the radiation events was caused by a load drop or slip, and none were significant.

There have been 30 crane events involving either a fuel assembly drop or damage to a fuel assembly caused by handling. However, given the steady increase in the number of operating units to more than 100 during the period of the survey, there was an overall improvement in time in fuel handling performance. From a risk perspective, none of the 30 fuel assembly drop or fuel handling events resulted in radiation exposure or risk to personnel.

There were few load slips involving very heavy loads. Of the estimated 54000 very heavy load lifts at operating facilities following the issuance of NUREG-0612 in 1980, there were six very heavy load slips. None of the six very heavy load events resulted in radiation releases, risks to licensee personnel or the public. In 1999, Comanche Peak Unit 1 had the most significant very heavy load slip event involving the slip of a reactor coolant pump motor of 4.6 to 6.1 meters (15 to 20 feet). As the load was rapidly falling, one link of the hoist chain randomly lodged in the lower chain block which arrested the unplanned descent. The motor stopped approximately 2.4 meters (eight feet) above the pump base. Had the link not lodged in the chain block, the motor could have continued dropping, damaging the reactor coolant pump and piping. The issue was determined to be of very low safety significance in the reactor safety strategic area because all fuel had been transferred to the spent fuel pool prior to the load slip. However, at the time of the slip, load control procedures allowed performance of this very heavy load lift in operational modes 5 or 6, where fuel would be present in the reactor vessel. Damage to reactor coolant system integrity in modes 5 or 6 significantly increases the probability of fuel damage because mitigating equipment necessary to recirculate lost coolant is not required to be available.

There were few load drops involving very heavy loads. Of the estimated 54000 very heavy load lifts at operating plants since the issuance of NUREG-0612, three very heavy load drops were identified. These three very heavy load drop events occurred because of human error, and ultimately because of rigging deficiencies and not because of crane deficiencies. The three events also did not occur near any safety related areas, and none resulted in radiation releases, risks to licensee personnel, or the public. The very heavy load drop event at Byron occurred while operating a mobile crane, while the San Onofre 3 and Turkey Point 4 very heavy load drop events occurred while operating turbine building overhead cranes.

Estimates of load handling failure rates are low. Based on actual crane operating experience data from commercial U.S. nuclear power plants, this study estimates the rate of load drops per demand for very heavy loads to be 5.6E-05. This estimate is an industry average, and may be higher or lower at a given facility because of varying human error rates which appear to dominate load drop events. NUREG-0612, which based its estimates on the data collected from the Navy, estimated the probability of a handling system failure for nuclear plant cranes will be between 1E-05 and 1.5E-04 per lift. This probability of failure was an estimate since Navy crane data does not indicate how many lifts were actually performed, i.e., only the number of problems has been quantified. A report issued by the Environmental Evaluation Group (EEG) of New Mexico, estimated the probability of failure of the TRUDOCK crane system at the Waste Isolation Pilot Plant (WIPP), to have a combined equipment failure rate per demand of 5E-06, and a combined operator error rate of 1.7E-07 per demand.

Although single-failure-proof cranes share many common design features (e.g., dual reeving, redundant limit switches, and redundant brakes), the remaining criteria for declaring a crane as single-failure-proof (e.g., for new cranes or upgraded cranes) have been inconsistently applied. Crane manufacturers have also stressed that NUREG-0554 is ambiguous in some areas, and that clarifications or changes need to be made to both NUREG-0612 and NUREG-0554. Industry suggested that a preferred approach would be to consider adopting ASME NOG-1, Type I (with minor changes) as an acceptable approach to meeting NUREG-0554 and for upgrading cranes to single-failure-proof status. NOG-1 contains much more specific design criteria for single-failure-proof cranes than does NUREG-0554. In addition, while some licensees listed a crane as single-failure-proof, or indicated that it met NUREG-0612 upgrade requirements, all the single-failure-proof design criteria listed in NUREG-0554 still may not be fully met. Among events occurring during the period 1968 through 2002 involving cranes suitable for an upgrade to a single-failure-proof design, most load drop events have been the result of poor program implementation or human performance errors that led to hoist wire rope or below-the-hook failures. All three very heavy load drops were the result of rigging failures, not crane failures. Consequently, there were no very heavy load drop events that could have been prevented had only a single-failure-proof crane been employed in the lift. However, there were load or hook and block assembly drops that could have been prevented with the use of single-failure-proof cranes and lifting devices.

The accuracy and consistency of information received by the NRC in response to generic communications (GCs) regarding load handling are questionable. There have been 29 NRC GCs that have involved load movements at U.S. nuclear power plants dating back to 1976. There have been nine GCs which discuss heavy loads moved on the refueling floor, load drop analysis for heavy loads, identification of heavy loads that are lifted over safe shutdown equipment, and the consequence of a load drop on selected equipment. A few GCs (issued as generic letters and one bulletin) requested licensees to provide information on their crane programs for NRC evaluation. Many of the licensees that responded to the latest information request (Bulletin 96-02), provided incomplete information. Also, in many instances, information previously provided to the NRC was not verified to be accurate.

Although not required by NRC regulations, few licensees have performed a consequence analysis of heavy load drops. Of the 74 facilities that responded to Bulletin 96-02 (requesting licensees to provide the staff with specific information relating to their heavy load programs and plans), eight licensees indicated that a consequence analysis had been done at their facility for heavy load drops.

The number of crane-related injuries has increased during the last decade. There have been 16 reported injuries involving crane operation during the period from 1969 through 2002. When comparing the last decade (1993-2002) with the second decade (1981-1992), a 100 percent increase in the number of injuries occurred concurrently with a 9 percent increase in the number of operating power plants.

Deaths caused by crane operation occurred largely during the construction phase of the nuclear power industry. There have been 10 reported crane events that have led to deaths in the nuclear industry for the period 1969 through 2002. The highest concentration of crane related deaths at nuclear power plants occurred during the first decade (1969 to 1980). For the first decade, six of eight events that led to a death occurred at facilities still under construction. The last death in a crane related accident in the U.S. nuclear industry was 1985.

FOREWORD

This report provides an in-depth review and analysis of crane operating experience at U.S. nuclear power plants. In support of NRC Strategic and Performance Goals, the operational experience and human performance insights contained in this report can be used to enhance the control of heavy loads to reduce the likelihood of crane accidents, particularly those that have the potential to release radioactive material, and will be used as a technical basis for recommendations that may initiate changes to NRC regulatory requirements, programmatic controls, or evaluations of heavy load movements at U.S. nuclear power plants in response to Generic Issue 186, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants.” This report was conducted by the Office of Nuclear Regulatory Research and also supports a variety of risk-informed activities performed at both operating plants, decommissioned plants, and at waste facilities and repositories. While this report does not provide plant-specific quantification of risk of heavy load drops in various plant locations, it does provide equipment failure rates and probabilities for load drops in general, and for very heavy load drops (for those loads greater than 27.2 metric tons (30 tons).

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ACKNOWLEDGMENTS

This report benefitted greatly through the willingness of several staffs at Brown's Ferry, Comanche Peak, Diablo Canyon, Dresden, Grand Gulf, Limerick, Oconee, Oyster Creek, and Palo Verde nuclear power facilities, in researching, documenting, and sharing crane operating experience information. Appreciation is also expressed to those individuals in industry, the public, and the U.S. Nuclear Regulatory Commission who reviewed this report for accuracy.

ABBREVIATIONS

ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
AE	architect engineer
AIT	Augmented Inspection Team
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
B&W	Babcock and Wilcox
BL	bulletin (NRC)
BWR	boiling water reactor
EDG	emergency diesel generator
CCDP	conditional core damage probability
CE	Combustion Engineering
CFR	Code of Federal Regulations
CMAA	Crane Manufacturers Association of America
COMP	component
CR	circular (NRC)
CRDM	control rod drive mechanism
DIST	slip or drop distance
DOE	U.S. Department of Energy
EEG	Environmental Evaluation Group
EQE	Engineering consulting firm
FSAR	final safety analysis report
GC	generic communication
GE	General Electric
GI	generic issue
GL	generic letter (NRC)
HEPA	high efficiency particulate air
HT	height
HVAC	heating, ventilation, and air conditioning
IGSCC	intergranular stress corrosion cracking
IMIS	integrated management information system
IN	information notice (NRC)
ISFSI	independent spent fuel storage installation
kg	kilogram
kV	kilovolt

ABBREVIATIONS (Continued)

MWt	megawatt thermal
NEI	Nuclear Energy Institute
NOG	nuclear overhead and gantry
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NSSS	nuclear steam system supplier
NUDOCS	nuclear documents system
OMDS	Office of Management Data Services
OSHA	Occupational Safety and Health Administration
PG&E	Pacific Gas and Electric
PWR	pressurized water reactor
RC	reinforced concrete
RCP	reactor coolant pump
RES	Office of Nuclear Regulatory Research (NRC)
RHR	residual heat removal
RPV	reactor pressure vessel
RWCU	reactor water clean up
S&L	Sargent and Lundy
SFP	spent fuel pool
SSE	safe shutdown equipment
SWEC	Stone and Webster Engineering Company
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
USI	unresolved safety issue
VHL	very heavy load
WIPP	Waste Isolation Pilot Plant
WT	weight

1 INTRODUCTION

1.1 Background

In nuclear plant operation, maintenance and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop they could impact on stored spent fuel, fuel in the core, or on equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In some instances, load drops at specific times and locations, could potentially lead to offsite doses that exceed 10 CFR Part 100 limits.

In April 1999, a candidate generic issue (GI) was proposed (Ref. 1) by the Office of Nuclear Reactor Regulation (NRR) of the U.S. Nuclear Regulatory Commission (NRC). NRR requested the Office of Nuclear Regulatory Research (RES) within the NRC to evaluate the issue. NRR was concerned that although licensees may be operating within the regulatory guidelines in Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants NUREG-0612," they may not be taking action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In other words, licensees may not be taking adequate measures, if any, to assess and mitigate the consequences of dropped heavy loads.

In May 1999, RES informed NRR (Ref. 2) that the candidate GI was accepted, and was given the title GI-186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Ref. 2 indicated that GI-186 would be prioritized in accordance with RES Office Letter No. 7, "Procedure for Identification, Prioritization, Resolution, and Tracking of Generic Issues." With the advent of Management Directive 6.4, "Generic Issue Program," in July 1999, it was decided to process this new issue in accordance with MD 6.4 instead of Office Letter No. 7.

1.2 Precursors to Initiation of Generic Issue 186

Several related events took place that led up to the initiation of GI-186. Significant related documents are discussed in chronological order.

- Unresolved Safety Issue (USI) A-36, "Control of Heavy Loads near Spent Fuel" (1970s)

  This issue focused mainly on potential consequences of a heavy load drop on fuel assemblies in either the spent fuel pool area or on the reactor that may result in; (1) a release of radioactivity because of a cladding breach, or (2) a critical mass of fuel in the core or in the spent fuel pool. USI A-36 was resolved with the issuance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and revisions to Section 9.1.5 of the Standard Review Plan, "Overhead Heavy Load Handling Systems."

- NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" (May 1979)

  NUREG-0554 was developed to provide design, installation, testing and quality assurance requirements for single-failure-proof cranes. The NRC has licensed reactors on the basis that the safe handling of critical loads can be accomplished by adding safety features to the handling equipment, by adding special features to the structures and areas over which the critical load is carried, or by a combination of the two. When reliance for the safe handling of critical loads is placed on the crane system itself, the system should be designed so that a single failure will not result in the loss of the capability of the system itself, the system should be designed so that a single failure will not result in the loss of the capability of the system to safely retain the load. This document (Ref. 3) identifies features of the design, fabrication, installation, inspection, testing, and operations of single-failure-proof overhead crane handling systems (limited to the hoisting system and to braking systems for trolley and bridge).

- NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (July 1980)

  This report (Ref. 4) provides the results of the review of the handling of heavy loads and includes the task group's recommendations on actions that should be taken to assure safe handling of heavy loads. This report completed Task A-36 described earlier. Subsequent documentation divided the NUREG action items into what became known as Phase I (Section 5.1.1) and Phase II (Sections 5.1.2 through 5.1.6). Phase I addresses safe load paths, procedures, crane operator training, special lifting devices, lifting devises that are not specially designed, and crane inspection and maintenance, while Phase II addresses alternative design requirements for cranes located in the spent fuel pool area for pressurized water reactors (PWRs), the containment building for PWRs, the reactor building for boiling water reactors (BWRs), and in other plant areas for either a PWR or BWR.

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

  This GL indicated that (1) all licensees had completed the requirement to perform a review and submit a Phase I and a Phase II report, (2) based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action was not required to reduce the risks associated with the handling of heavy loads, (3) a cost-benefit analysis of PWR polar crane conversion to single-failure-proof was not cost beneficial, and (4) a detailed Phase II review of heavy loads was not necessary and that Phase II was considered completed.

- Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel in the Reactor Core, or Over Safety-Related Equipment" (April 1996)

  This bulletin (BL) was initiated because of load drop analysis performed by the Oyster Creek nuclear power plant. The BL: (1) alerted licensees to the importance of complying with existing regulatory guidelines on the control and handling of heavy loads, (2) reminded licensees of their responsibilities for providing adequate protection of public health and safety when handling heavy loads during plant operation, and (3) alerted licensees to the potentially high consequences that may result from a cask drop, and the importance of taking measures to mitigate such consequences in addition to measures to preclude the load drops.

- There have been 29 NRC generic communications have been issued since the 1970s concerning fuel handling and crane operating experience. Many have to do with load handling programs, fuel assembly drops and damage, load paths, load drop analyses, and requests for information. These generic communications are summarized in Appendix G, “NRC Generic Communications Involving Crane Operating Experience.”

- Generic safety issue proposed by NRR, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants” (April 1999)

  NRR had previously studied the issue as part of the “Dry Cask Storage Action Plan,” and later as the Heavy Load Control (HLC) and Crane Issues Task Action Plan” prior to requesting assistance from RES.

2 CRANE OPERATING EXPERIENCE AT U.S. NUCLEAR POWER PLANTS

The Nuclear Documents System (NUDOCS) database and the NRC’s Agencywide Access and Management System (ADAMS) were searched for documents relating to cranes for the period 1968 through 2002. Given the time period, crane events recorded included those occurring during construction and operation, and in some instances, during decommissioning. Each crane related document was reviewed and critical information was entered into a database for further analysis.

2.1 Basic Crane Program Requirements for Nuclear Power Plants

Noncompliance with accepted crane operating good practices, designed to reduce the likelihood of a major crane accident effecting the power plant or the public, was viewed as a major contributor to current or future crane accidents. For this reason, guidance provided in NUREG-0612 was reviewed. Phase I of NUREG-0612 includes preventative measures in the form of programmatic and human factors practices for heavy loads handled in the area of the reactor vessel or spent fuel in the spent fuel pool.

According to the NUREG-0612, 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) <u>Safe load paths</u> 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) <u>Procedures</u> 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. These procedures should include (see also Table 3-1 of NUREG-0612):

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 Section 3.3.1.1 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 3.3.1.1 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 operations. 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.)

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

2.2 Crane Event Database Categories and Subcategories

To analyze crane issues, several general categories were established to help determine the causes of crane problems and their effect or consequence. This information was saved in a database, and sorts were performed to discern trends and patterns. Many crane event reports lack detailed information on causes, load description, or crane type. Given the cursory information that was available on many events, basic information, such as event date (pre- or post- NUREG-0612), crane type, and whether Phase I criteria of NUREG-0612 were adequately implemented.

Table 1: Crane event database categories and subcategories

General Event Category	Event Subcategories
Plant and event date	Docket, plant name, event year, event month, operation date, shutdown date, whether the issue occurred following the issuance of an operating license and after issuance of NUREG-0612, single-failure-proof crane for cask movement
Crane type	Reactor building, polar, auxiliary building, refueling/manipulator, spent fuel pool, tower, mobile, other
Crane component deficiency	Structure, control, brakes, rails, fasteners, component failure, below-the-hook, unknown, none
Reported cause of event	Not following procedures, poor procedures, test performance, load path inadequacy, ventilation inadequacy, maintenance, engineering, operations, unknown, none
Safety Implication of event	Death, injury, radiation release, load slip, load drop, equipment damage, crane component drop, loss or partial loss of power, fuel drop or damage, none
Event abstract	Event description (component and weight), distance of load of drop or load slip

2.3 Analysis of Crane Events at Nuclear Power Plants

A review of crane documents for the period 1968 through 2002 resulted in 430 different issues involving mostly large capacity cranes used at nuclear power plants during construction and operation. Events involving small hoists were not included in this survey. Crane issues were reported by individual licensees, through NRC documents and inspection reports, by vendors, and the public. It is assumed that many minor crane events were not reported in licensee corrective action programs, licensee event reports (LERs), or NRC inspection reports. It is also assumed that minor crane or hoist events were not always reported, but that major crane events that involved a drop, slip, or damage did get documented. Most reported issues were the result of programmatic implementation weaknesses (i.e., not following a procedure, inadequate load paths, noncompliance with technical specifications, inadequate testing prior to use, etc.). The following figures not only include a wide span of operating experience from construction to operation, but also include a wide variety of crane types, some of which are not used at operating nuclear facilities today. Figures 1 through 18 present nuclear crane operating experience as a whole regardless of the weight of the load being lifted, or whether the lift was done during construction, during an outage, or during plant operation. Section 3.0 of this report discusses a subset of information contained in this section, in that it contains an analysis of crane operating experience involving “very heavy loads” (e.g., loads greater than approximately 27.2 metric tons [30 tons]) at nuclear power plants that have received an operating license.

2.3.1 Reported Crane Issues

Figure 1, “Reported crane issues at U.S. nuclear power plants,” shows the total number of reported crane issues in two-year increments. Although there has been a steady increase in the number of crane issue reports since the late 1960s, when compared to the number of operating units as shown in the figure, an improving trend (based on issues reported per number of operating units) is noticed. The two-year period 1997 through 1998 appears to be an outlier compared to operating experience data documented prior to and after that time period. The 1997 through 1998 period also experienced a high percentage of human errors (83 percent), including a high number of load drops (8).

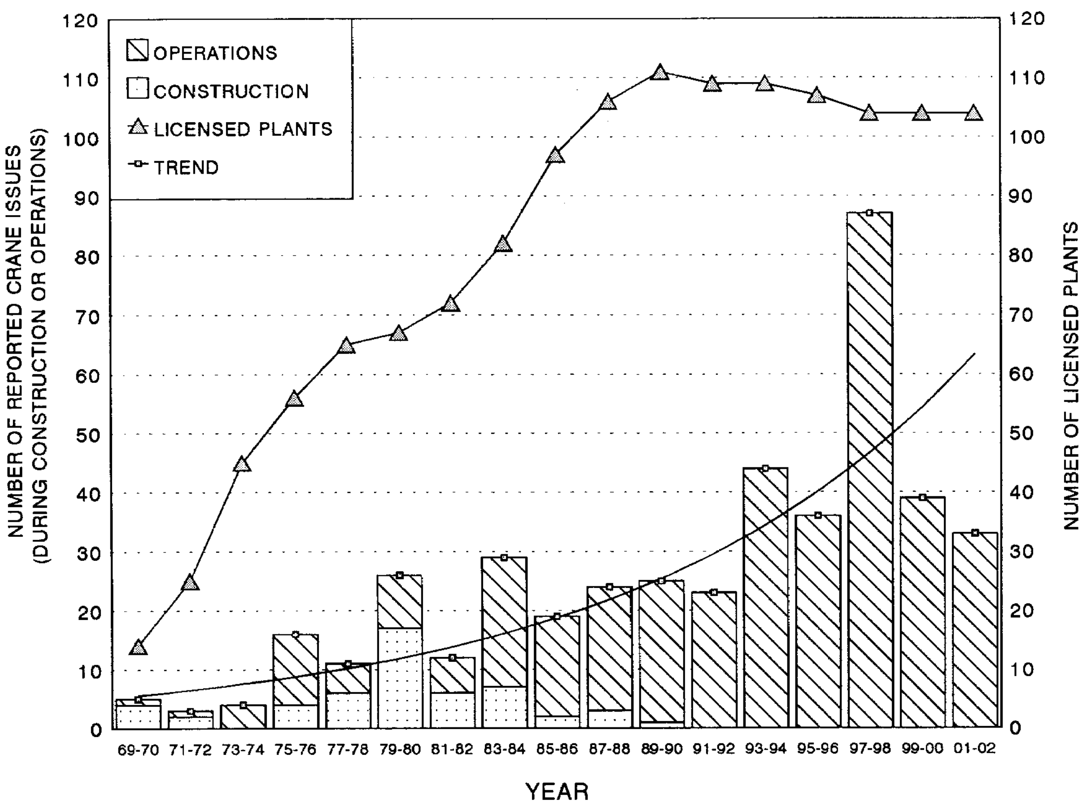

Figure 1: Reported Crane issues at U.S. nuclear power plants

2.3.2 Crane Reports Due to Poor Human Performance

Figure 2, "Trend in crane issues due to poor human performance," shows that poor program implementation was a major contributor to crane performance. Examples of poor program implementation include failure to perform surveillance tests, not following procedures, load path violations, and not obtaining necessary plant conditions prior to load movements. Figure 2 includes all documented issues related to crane operating experience regardless of the weight of the load being lifted. As shown in the figure, the percentage of crane issue reports caused by poor human performance has increased over time, and for the last several years, averaged between 70 and 80 percent. The line shown in the figure is a best fit trend line. The average percentage of crane issue reports caused by poor human performance for the entire time period (1969 through 2002) was calculated to be 73 percent. Similarly, when considering only very heavy loads (e.g., loads greater than 27.2 metric tons [30 tons]), the percentage of crane issue reports caused by poor human performance is 56 percent. Potential reasons for the reduction in error rate for very heavy loads could be the increased level of attention, extent of pre-job briefings, operator training, and operator experience of those associated with very heavy load lifts. In either instance, however, a significant reduction in the number and severity of crane events could be achieved through greater adherence to existing program guidance.

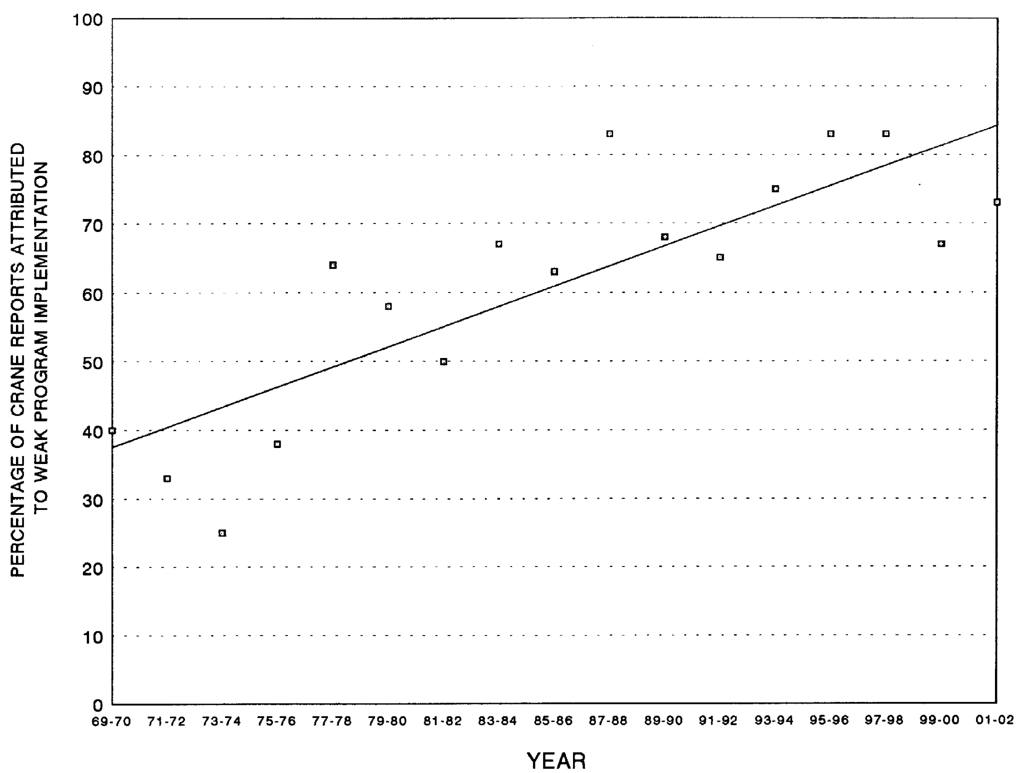

Figure 2: Trend in crane issues due to poor human performance

2.3.3 Crane Event Distribution by Crane Type

For the 430 reported crane issues during the period 1968 through 2002, Figure 3, “Crane issue distribution by crane type,” shows the total number of crane issues documented for each crane type. The number of crane issues for each crane type was not broken down by reactor type, operational phase, or weight of load at the time of the event. Crane types include polar, spent fuel pool (SFP), tower, auxiliary building, refueling/manipulator (MC), reactor building (RB), mobile, and other. The category “other” refers to cranes which do not specifically fit into one of the remaining categories, and could include turbine building cranes, special cask handling cranes, unspecified cranes, or miscellaneous cranes used inside or outside of areas containing safety-related components. If the event report did not specify the crane type, it was categorized as other. The crane reporting the most number of issues (approximately 28 percent of the total) was the SFP crane (121). The reason for the high number of issues is due in part to the number of times that the crane is used in moving fuel assemblies. The SFP cranes were responsible for many procedural violations involving building ventilation requirements, and fuel assembly drop events. The “other” category is high (100) because it represents several miscellaneous crane types. Small cranes and hoists were not included as part of this survey.

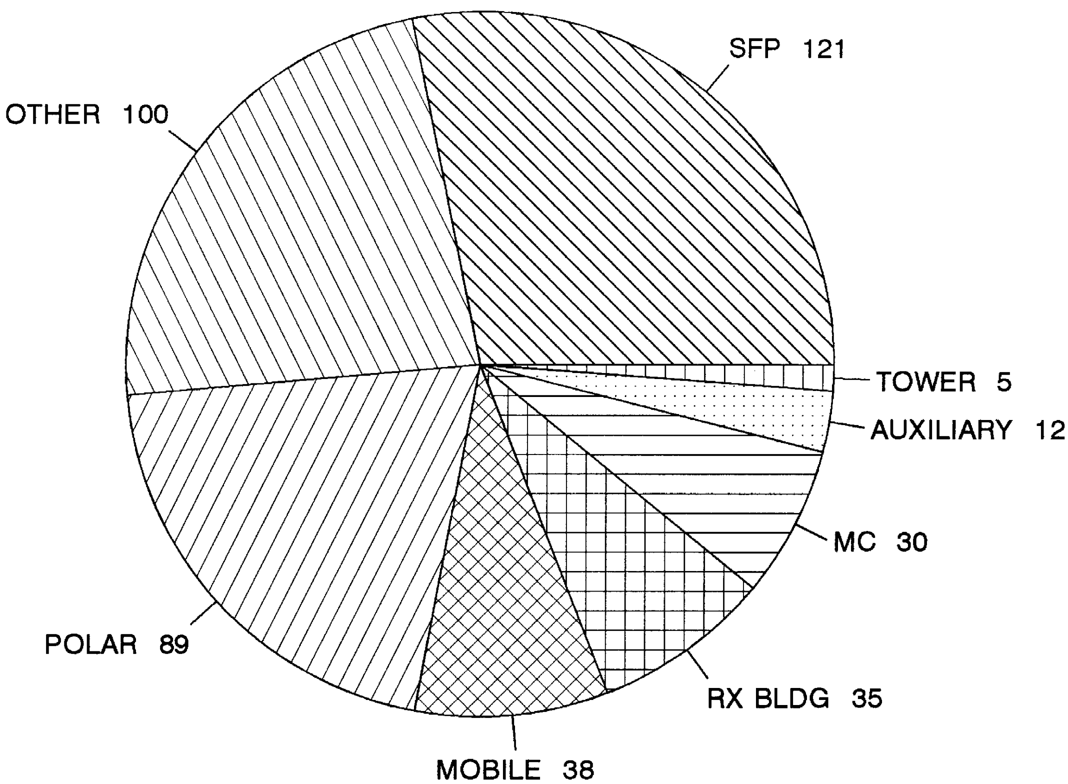

Figure 3: Crane issue distribution by crane type

2.3.4 Crane Types Involved in Load Drops, Load Slips, and Crane Component Drops

Figure 4, "Crane types involved in load drops, load slips, and crane component drops," shows the total number of significant crane events such as a load drop, a load slip and a crane component drop. A load drop is defined as an uncontrolled lowering of a load to the point where contact with the floor or some object stops any further decent. A load slip is an uncontrolled vertical movement of a load that appears to be intermittent. A crane component drop is the drop of some crane component such as the hook or other component different from the load itself. As shown by the figure, load slips, which are generally the result of hoist control system deficiencies, are much less prevalent than load drops which can result from either hoist failures or below-the-hook (mostly rigging) failures. Load drops while operating mobile and other cranes (representing almost half of the events) have been outside of safety related areas. Load drops while operating the SFP crane (representing over half of the load drop events) were largely because of fuel assembly drops which has posed no safety issue. For load drops involving polar or RB cranes, four involved fuel assembly drops, two involved an auxiliary hoist on the RB crane and were not classified as very heavy loads, one involved a rigging failure of no consequence, and one occurred during early plant construction.

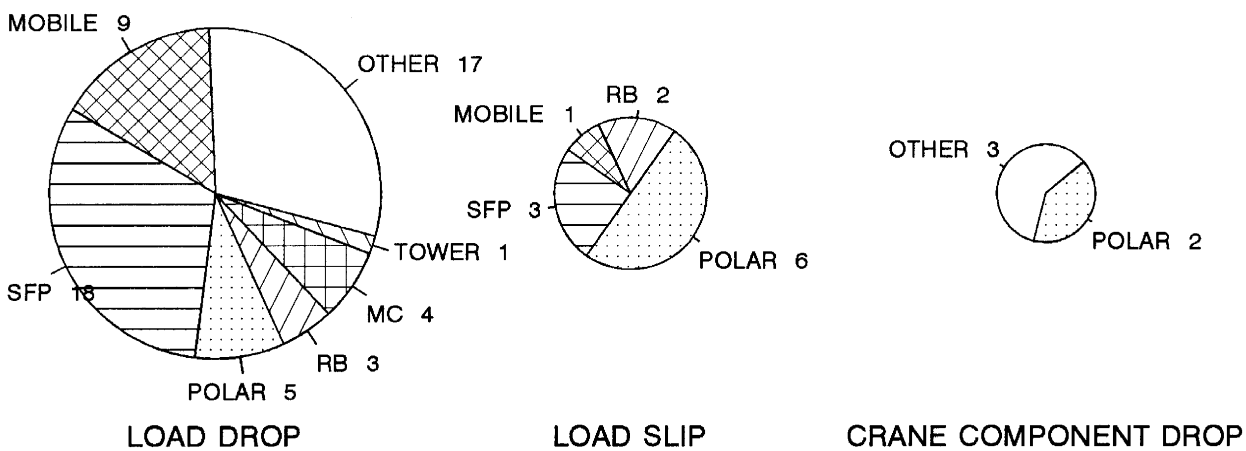

Figure 4: Crane types involved in load drops, load slips, and crane component drops

2.3.5 Crane Events Due to Hardware Deficiencies

As shown in Figure 5, "Crane events due to hardware deficiencies," of the 430 crane issues, 156 (approximately 36 percent) involved crane equipment or hardware problems. The crane issue was assigned to the category "Unknown" if a malfunction had clearly occurred, but the document did not indicate what component had failed. The crane issue was assigned to the category "Components" if the component that failed did not specifically fit into the remaining hardware categories. In addition to the 156 crane issues, there were 47 "Below-the-Hook" events involving mostly rigging failures or deficiencies not associated with the crane itself. The most prevalent hardware issue involved control system issues. Control system issues resulted in four load slips and 12 load drops, none of which involved single-failure proof cranes. In addition, there were many other control system issues involving load cells, limit switches. As can be seen from the figure, 236 of 430 reported crane issues did not involve any crane hardware deficiency, but were largely administrative or testing related.

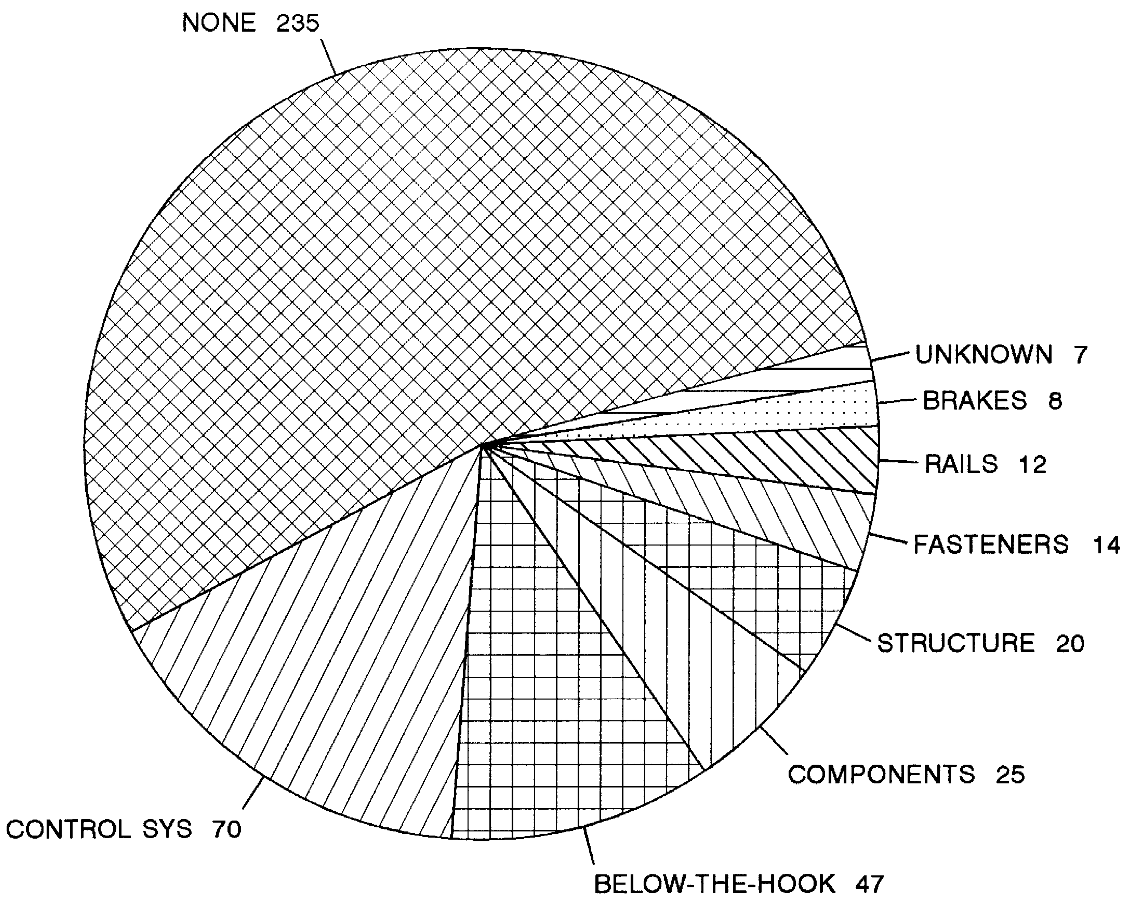

Figure 5: Crane events due to hardware deficiencies

2.3.6 Crane Events Caused by Weak Program Implementation

Upon review of the 430 crane issues, a cause of the issue was either listed in the crane issue report, was determined by the available facts presented in the document, was indeterminate (category “Unknown”), or there was insufficient information given in the report to conclude that any deficiency existed (category “None”). Figure 6, “Principal reasons for crane events,” shows the distribution of causes for the crane issue being reported. “Not Following Procedures” was the largest category with 159. Other categories that are similar to “Not Following Procedures” would be “Ventilation” (i.e., failure to establish the required ventilation prior to load movements in certain areas), “Did Not Test” (i.e., failure to perform crane surveillance tests prior to use) and “Load Path” (i.e., failure to move loads over established safe load path areas). There were 61 crane events where the cause of the event was categorized as “Unknown.” In most of these instances, the event report lacked sufficient detail to determine the cause with certainty.

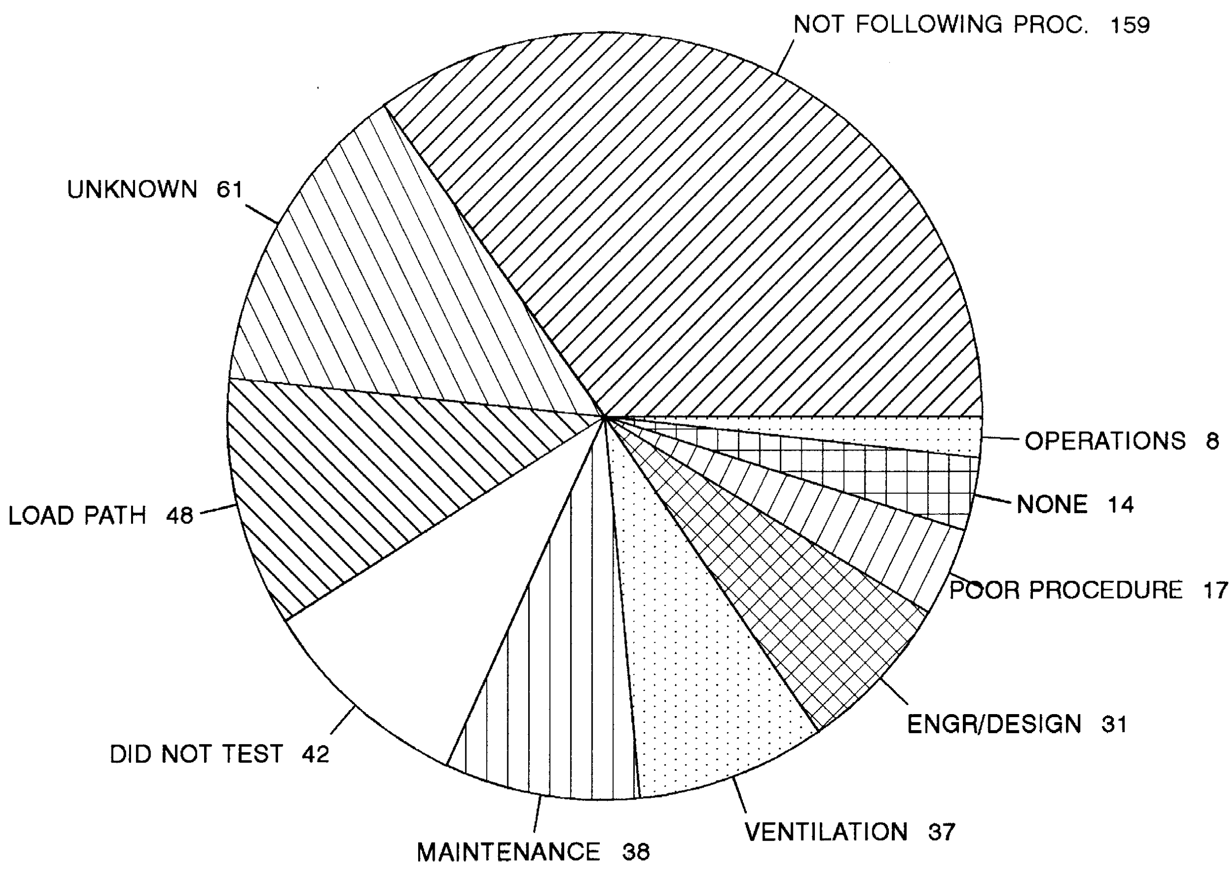

Figure 6: Principal reasons for crane events

2.3.7 Safety Implication of Crane Events

To assess the potential safety impact of reported crane issues, it was decided to review the outcome of each event or issue, and then to assess the outcome. Several outcome categories were established: (1) Death, (2) Injury, (3) Radiation Exposure, (4) Load Slip, (5) Load Drop, (6) Equipment Damage (i.e., equipment damaged by crane operation, or the crane itself was damaged during operation, (7) Loss or Partial Loss of Power, (8) Crane Component Drop (i.e., miscellaneous components falling from the crane), (7) Fuel drop/damage, and (8) None (i.e., no impact on plant equipment, workers or the public). In addition, within each category, multiple events may occur such as multiple equipment damage or injuries. Figure 7, "Safety effect of crane events," indicates the number of crane issues or events for each category, and not the quantity of items affected for each event. Consequently, of the 430 crane issues, the total number of "outcomes" is 531. As shown in Figure 7, approximately half of the crane issues resulted in no impact to plant equipment, workers or the public (e.g., the "None" category). The second largest contributor, "Equipment Issue," relates to either crane equipment malfunctions, crane damage, or to damage done by the crane during the event to other equipment. Most equipment damage was minor and of no consequence. Section 2.3.11 provides crane event details for those events which resulted in deaths or injuries. The were three crane events that resulted in radiation exposures, each were caused by human error. At the Pilgrim facility in 1979, a crane operator lifted an irradiated fuel assembly out of the spent fuel pool, resulting in increased exposure. At Turkey Point Unit 3 in 1992, a maintenance person was inattentive during movement of the polar crane, and fell into the refueling cavity and got contaminated. A third radiation event caused by crane operation occurred at Farley Unit 2 in 1999 when the failure of the polar crane primary height measuring system allowed a portion of the reactor lower internals to be exposed during a lift. None of the radiation events was caused by a load drop or slip, and none were significant.

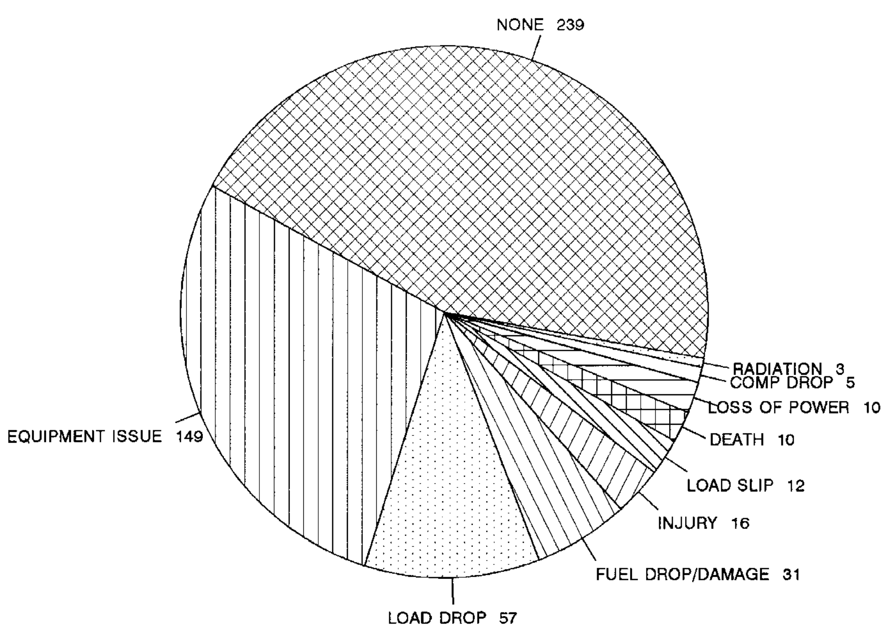

Figure 7: Safety effect of crane events

2.3.8 Crane Events Involving a Load Slip

During the period 1968-2002, there were 12 reported events involving load slips. Figure 8, "Load slip distribution," shows that there has been an increase in the number of load slips within the last decade (1993-2002) during a period where the number of operating nuclear power plants has remained fairly constant. No load slip events were recorded which involved plants that were still in their construction phase, probably because of the lack of reporting during that time period. No one cause dominated the events. Load slips were caused by below-the-hook issues, control systems, operations and engineering. Six of the 12 load slip events occurred while operating a polar crane, however, most of these events were beyond the control of the crane or the crane operator.

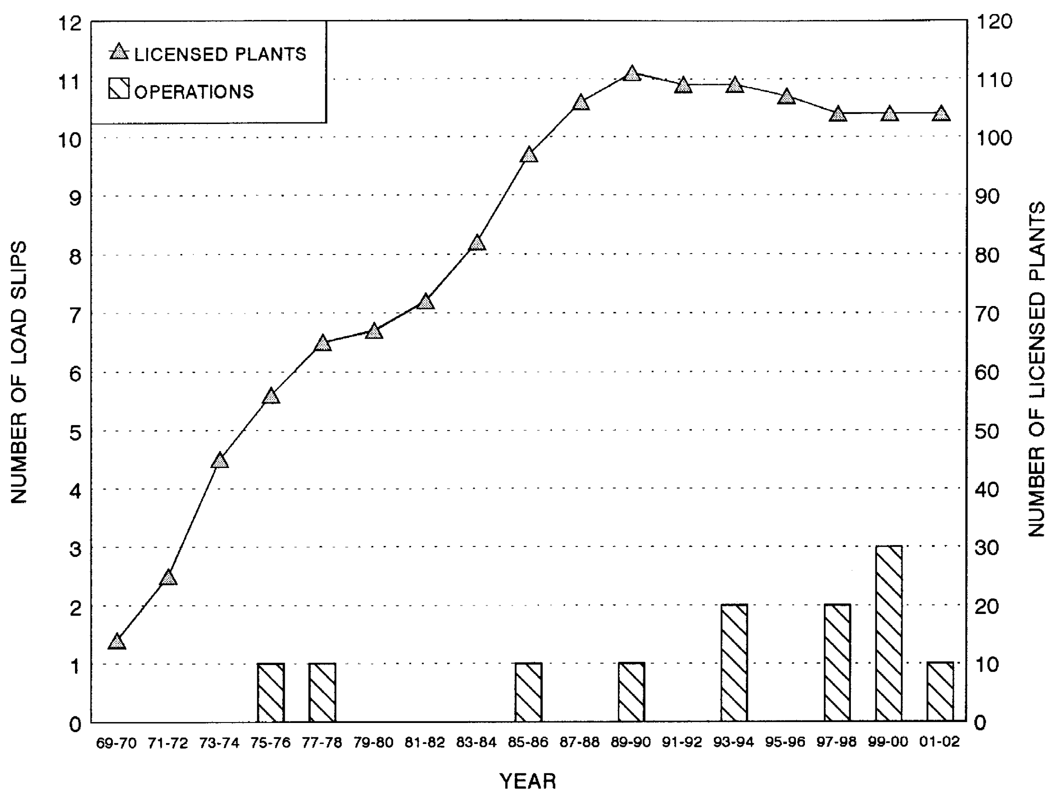

Figure 8: Load slip distribution

The most significant load slip event occurred at Comanche Peak Unit 1 in 1999, and involved the lift of a reactor coolant pump motor weighing approximately 38.1 metric tons (42 tons). Removal of the motor was performed using a nonsingle-failure-proof 40.8 metric tons (45 tons) electric-driven chain hoist that was rigged to the polar crane main hook which was rated at 158.7 metric tons (175 tons). Use of the chain hoist was necessary to raise the motor approximately 24 meters (80 feet) from its base because the main hook of the polar crane was too large to be lowered into the narrow compartment above the pump. While raising the motor, the auxiliary hoist gearbox failed, allowing the chain to free-wheel through the chain blocks and the motor dropped. As the load was rapidly falling, one link of the hoist chain randomly lodged in the lower chain block which arrested the unplanned descent. The motor stopped approximately 2.4 meters (eight feet) above the pump base. Had the link not lodged in the chain block, the motor could have continued dropping, damaging the reactor coolant pump and piping. The issue was of very low safety significance in the reactor safety strategic area because all fuel had been transferred to the spent fuel pool prior to the load slip. However, at the time of the slip, load control procedures allowed performance of this very heavy load lift in operational modes 5 or 6, where fuel would be present in the reactor vessel. Damage to reactor coolant system integrity in these modes significantly increase the probability of fuel damage because mitigating equipment necessary to recirculate lost coolant is not required to be available. Had this event occurred with fuel in the reactor vessel, and the chain had not jammed in the hoist, this event could have had significantly greater safety significance. For brief summaries of other load slips see Table 2, "Reported crane events involving a load drop or a load slip."

2.3.9 Crane Events Involving a Load Drop

During the period 1968-2002, there were 57 reported events involving load drops. Figure 9, "Load drop distribution," shows the event distribution, a straight line representing the best fit curve for the 34 year period, and the number of operating nuclear plants. Table 2 provides a brief description of each event. Load drops while operating the spent fuel pool crane (representing over half of the load drop events) were largely because of fuel assembly drops caused by grapple operation or human errors which posed no safety issue. Load drops while operating mobile and other cranes (representing almost half of the events) have occurred outside of safety related areas. However, several load drops have involved overhead cranes similar to those used in safety-related areas of the power plant. A comparison of approximately three decades (i.e., 1969 to 1980, 1981 to 1992, and 1993 through 2002) of operating experience involving load drops provides the following observations.

Decade 1. There were 18 events, of which 8 occurred during plant construction (prior to receiving an operating license). These 18 events also occurred during a period of new operating units coming on line as shown in the figure, but during a period of relatively few operating units. The reduced number of operating plants would indicate a higher drop incidence rate during this decade.

Decade 2. There were 15 events, of which 2 occurred during plant construction. This decade is characterized by a significant increase in the number of operating power plants accompanied by a slight decrease in the number of load drop events.

Decade 3. There were 24 events, of which none involved construction facilities. This decade is characterized by a leveling off of the number of operating facilities, but an increase in the number of load drop events. When compared to Decade 2, Decade 3 experienced a 60 percent increase in the number of load drop events, given a nine percent increase in the average number of operating units.

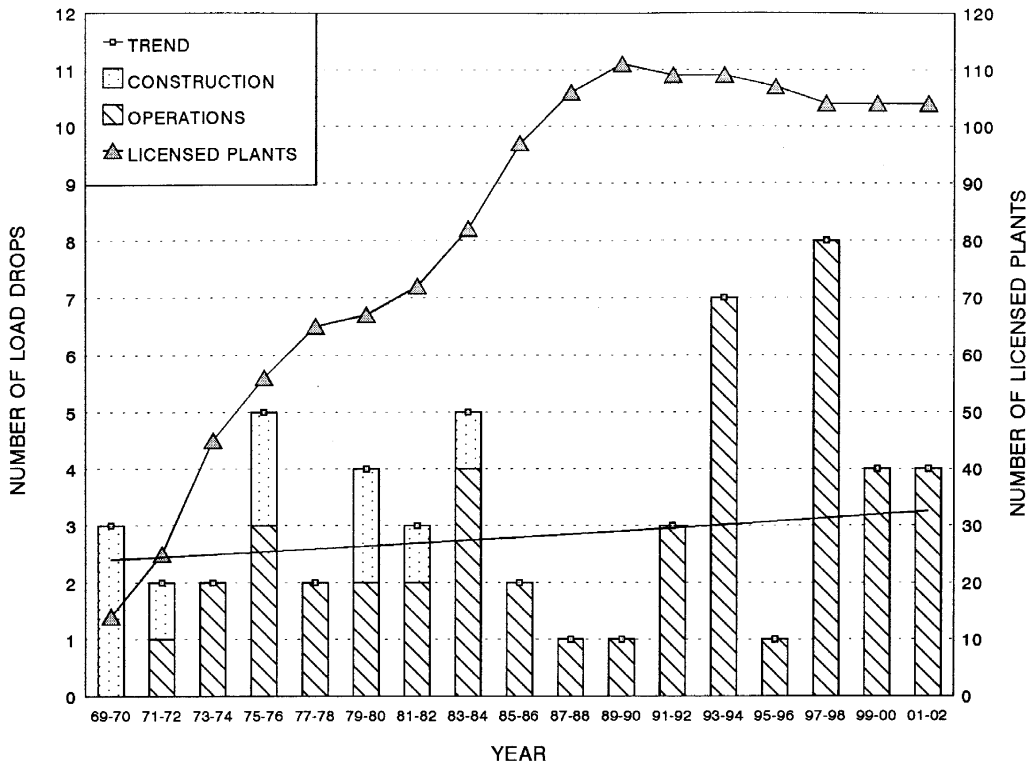

Figure 9: Load drop distribution

2.3.10 Load Drop Incidence Rate

Figure 10, "Load drop incidence rate," represents the total number of reported load drops divided by the accumulated number of operating reactor years for the period 1969 through 2002. The number of load drops includes those events occurring at both construction, shutdown, and operating facilities. The figure shows two curves, (1) one including all load drops regardless of the weight of the load, and (2) one including load drops considered in this report as "very heavy" (e.g., load weights of approximately 27 metric tons [30 tons] or more). These two curves also represent events occurring during construction, during operational phases, in safety-related areas and in nonsafety-related areas.

Another ratio of interest would be load drops per lift, however, the number of lifts for all cranes and all load weights is unknown. The number of lifts for very heavy loads is known to be in excess of 54000. The slight increase in the very heavy load drop incidence rate during 2001 is the result of two similar below-the-hook load drop events (San Onofre in May 2001, and Turkey Point in June 2001). In each instance, a mobile crane weighing 34 metric tons (37.5 tons) was dropped by the turbine building overhead crane due to rigging failure. See Table 2 for additional information.

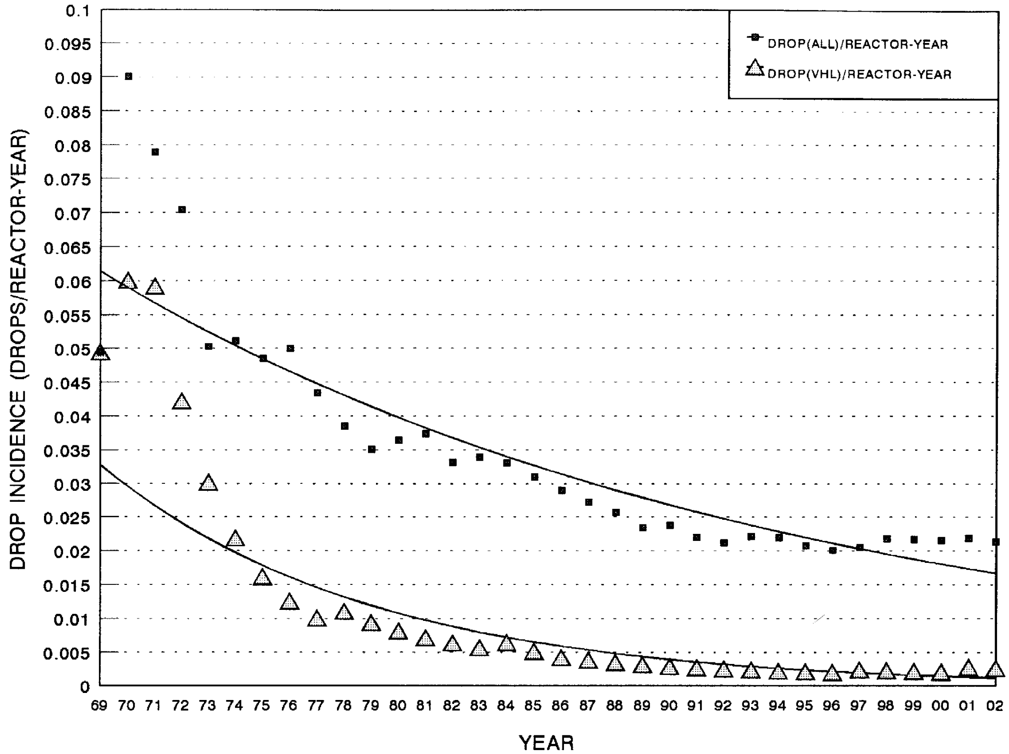

Figure 10: Load drop incidence rate

Table 2: Reported crane events involving a load drop or a load slip

Plant	Event Date	Event Type	Event Description
Ginna	July 1969	Load Drop	An assembly was dropped (due to a crane brake failure) which included the core barrel, the thermal shield, lower core plate and attached internals weighing about 82 metric tons (90 tons). The assembly was partially supported during its fall by the crane brake. The assembly tilted slightly as it fell approximately 1.8 meters (six feet) to a temporary storage support which acted as an energy absorber. Evaluation of the event indicated that the crane motor overheated, the electromagnetic brake failed and a backup mechanical brake was removed as part of a modification by Westinghouse.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Turkey Point 3	March 1970	Load Drop	A special crane erected on the turbine pedestal collapsed when two vertical support cables snapped while lifting the Unit 3 generator stator into its permanent location, killing one person and injuring two others.
Palisades	Sept. 1970	Load Drop	A cable on a 23 metric tons (25 ton) auxiliary crane broke during a transfer of a control rod drive mechanism (CRDM) support tube from the reactor vessel head area to a disassembly area inside containment. The broken cable allowed the CRDM support tube, including the crane block and hook to fall (weighing approximately 953 kg [2100 pounds]) approximately 6.7 meters (22 feet) to the reactor vessel head. The crane operator bypassed the upper limit electrical interlock and drove the crane sheave into the mechanical stop, breaking the crane cable. Visual damage appear to be limited to gouges on the flange surfaces of two CRDM housings, and bending of the dropped support tube.
Indian Point 3	Jan. 1971	Load Drop	The reactor vessel weighing approximately 402 metric tons (443 tons) underwent an unscheduled descent while it was being hoisted prior to its placement. It was not clear what caused the descent. Two failures occurred, (1) the crane cable, and (2) the pinion gear bracket to base plate welds on the hoist mechanism itself. The order of the failures was not known. The time of the descent was “certified” to be between 15 and 60 seconds. It was concluded that no damage to the pressure vessel occurred as a result of the incident.
Fermi 1	Oct. 1972	Load Drop	While transferring fuel from an auxiliary fuel storage facility to the Fuel and Repair Building, a crane operator inadvertently actuated the “raise” instead of the “lower” control, causing the 0.64 centimeters (1/4 inch) bolt in the shackle holding the subassembly to fail. As a result, the subassembly fell 8.2 meters (27 feet) into the transfer tank.
Pilgrim	Jan. 1974	Load Drop	An irradiated fuel assembly became detached from the grapple and fell in the spent fuel pool.
Millstone 1	Sept. 1974	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool.
Duane Arnold	June 1975	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool.
Humbolt Bay	June 1975	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool.
Brunswick 2	March 1976	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool. The assembly fell before it was fully inserted into its rack.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Brunswick 2	March 1976	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool. The assembly fell to a horizontal position across the top of the spent fuel pool storage racks.
Comanche Peak	May 1976	Load Drop	While lifting a personnel bucket (unoccupied) with a mobile crane, it became unbalanced. The crane boom failed, coming to rest in the turbine mat area.
Dresden 2,3	May 1976	Load Slip	The reactor building crane was being used to reinstall the Unit 2 reactor vessel head, using an "inching" motor. At one point, upon termination of downward drive, the head dropped abruptly approximately 38 centimeters (15 inches) before the brake engaged. A second abrupt drop was observed before the head was seated on the reactor. Both drops occurred as the head was being guided down over the reactor vessel studs, with thread protectors installed on four studs being used as guides. No forcible contact with the flange or studs occurred, and no damage resulted to either the crane or reactor components. Troubleshooting of the brake discovered sporadic arcing of new contacts at the time of inching motor drive termination. The inching motor portion of the recent modification was tagged out of service.
Comanche Peak	May 1976	Load Drop	While lifting a personnel bucket (unoccupied) with a mobile crane, it became unbalanced. The crane boom failed, coming to rest in the turbine mat area.
Peach Bottom 3	Jan. 1977	Load Drop	A fuel assembly was inadvertently released from the grapple and fell across the core. The cause was attributed to operation of the grapple open switch on operator error.
Oyster Creek	May 1977	Load Slip	A fuel assembly and mast dropped while lowering the assembly into the spent fuel pool racks. The drop was arrested by the cable drum brake. However, the slip resulted in shearing six bolts that coupled the refueling mast speed reducer to the cable drum. An examination indicated that 4 of the 6 bolts had failed at some earlier date.
Crystal River	June 1978	Load Drop	A missile shield crane hook failed, dropping its test weight on a fuel assembly causing minor damage. The crane hook was plant fabricated.
Pilgrim	Dec. 1979	Load Drop	A new fuel assembly was being moved to the spent fuel pool using the reactor building crane, when the assembly struck the top edge of the high density fuel racks and the latching device on the auxiliary hook failed to retain the fuel. The assembly fell, striking the lifting bails on four spent fuel elements, then coming to rest on the top of the fuel racks.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Three Mile Island	Feb. 1980	Load Drop	A mobile crane tipped over wile moving a load of scrap metal outside the gate. Specifics not available.
Hartsville	May 1980	Load Drop	While pouring concrete, the crane brake failed dumping concrete and severely injuring three workers.
Shearon Harris	May 1980	Load Drop	A crane inside containment lifted a lifting tackle approximately 43 meters (140 feet) when the binding broke. The lifting tackle fell, landed first on scaffolding, and then onto eight workers. Various injuries were received. The cause is unknown.
Prairie Island 1	Jan. 1981	Load Drop	The top nozzle and fuel assembly separated, which led to an assembly drop (weighing less than 0.9 metric ton [2000 pounds]). The break was caused by IGSCC.
Cook 1	June 1981	Load Drop	A fuel bundle was damaged while it was being transferred in the refueling cavity using the manipulator crane, when the lower end of the assembly struck a ledge on the refueling cavity floor just outside the reactor vessel area. One rod was dislodged and fell from the assembly onto the refueling cavity floor. No radiation was released.
Callaway	July 1981	Load Drop	A crane boom collapsed on the service water building while lifting concrete hatch weighing 11 to 16 metric tons (12 to 18 tons).
River Bend	March 1983	Load Drop	A 363 metric tons (400 tons) form assembly for the containment shield building roof was being lifted to the top of the cylindrical containment shield building, after which concrete would have been poured to form the shield roof. The day before, the 3.8 centimeters (1.5 inches) thick steel containment building dome had been successfully lifted and placed on the containment building by the same crane. When the form was about 9 meters (30 feet) above its assembly area and was about to be moved to position for lifting and placement on eh shield building, the crane mast buckled and the shield form fell to the ground and the crane collapsed. Except for the shield form, no permanent structures or equipment were damaged. Cause of the crane failure was not determined.
Turkey Point 4	April 1983	Load Drop	A fuel assembly was being inserted into the core. It was not aligned properly, and fell over so that is leaned at a 35 degree angle against two other fuel assemblies.
Turkey Point 4	April 1983	Load Drop	During a fuel bundle lift from its storage rack, the limit switches failed to stop upward movement. The hoist two blocked, parting the hoist cable and causing the assembly to drop back into its rack .

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Hatch 1	Oct. 1984	Load Drop	A possible inadvertent actuation of the fuel grapple hook position switch resulted in dropping a spent fuel bundle about 4 meters (12 feet) into its storage rack cell, slightly deforming and scratching the bundle and rack. No radiation release occurred.
Millstone 2	Nov. 1984	Load Drop	A spent fuel pin was dropped while performing eddy current testing. The cause was attributed to inadequate gripping force on the pin.
Brown's Ferry 2	March 1985	Hook Drop	A maintenance worker was killed and three others were injured when they were struck by a falling crane hook inside the unit turbine building. The accident occurred when the overhead crane cable parted. The 23 metric tons (25 tons) capacity hook dropped through the roof of a temporary building where the maintenance workers were located.
St. Lucie 1	Nov. 1985	Load Slip	While performing a lift of the upper guide structure weighing approximately 45 metric tons (50 tons), a bolt used to help secure a rigging device failed because it was improperly threaded. The upper guide structure tilted approximately 15 centimeters (6 inches) when the bolt failed. No damage was done.
Three Mile Island 2	Dec. 1985	Load Drop	While loading fuel assembly end fittings into a defueling canister, an end fitting became stuck in the canister. During attempts to reposition the stuck end fitting with the 0.9 metric ton (one ton) jib crane, the defueling canister and support sleeve were dislodged from the canister positioning system, and dropped. The canister and sleeve fell approximately 46 centimeters (18 inches) onto the top of the debris bed in the reactor vessel. The dropped load weight was 1.0 metric ton (2200 pounds), while the crane was rated at 0.9 metric ton (2000 pounds).
Haddam Neck	Feb. 1986	Load Drop	During the lift of the upper core support structure weighing approximately 26 metric tons (28.5 tons), a fuel assembly stuck to the structure because of a bent fuel assembly locating pin. The assembly fell off when the load was moved laterally. The dropped assembly and the two fuel bundles that it impacted were damaged, but there was no radiation release.
Grand Gulf	July 1987	Load Drop	A container of two new fuel bundles fell off a transfer cart to the turbine deck because of crane operator and rigging issues. Both fuel bundles had minor damage and were not used.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Quad Cities 1	Sept. 1989	Load Drop	During the transfer of new fuel from the new fuel storage vault to the fuel pool, a fuel assembly was released from the refueling grapple and fell upon the spent fuel racks. The grapple control switch was left in the “release” position when it was decided to lift the fuel to reposition it. The fuel was released, falling to the rack. The dropped fuel assembly, including the irradiated fuel it fell on, were visually examined in place from the bridge and the floor for signs of fuel damage. No damage was observed. Although no apparent damage resulted the fuel, 12 of the 32 potentially impacted fuel assemblies were discharged instead of reloaded for use in the next fuel cycle. The dropped fuel bundle was to be returned to GE.
North Anna 1	Jan. 1990	Load Drop	While the fuel building ventilation system was not aligned to discharge through the auxiliary building HEPA filter and charcoal absorber assembly, one fuel rod inadvertently slipped from the fuel rod handling tool due to a mechanical failure of the gripper mechanism, and dropped into its proper storage location in an uncontrolled manner. The height of the drop was not recorded, but no damage was recorded.
Fort Calhoun	April 1990	Load Slip	While lowering the reactor head, the load shifted, resulting in bending two alignment pins, and scratching the head flange.
Byron 2	Sept. 1990	Load Drop	A fuel assembly slipped out of the basket and dropped to the top of an empty fuel rack.
Indian Point 3	Oct. 1990	Load Drop	While lifting the upper core support structure weighing approximately 54 metric tons (60 tons), two fuel assemblies were found to be attached. One of the assemblies dropped into a retrieval basket when the brakes on the overhead crane were applied. A guide pin on each assembly was bent. The guide pins were most likely damaged during the previous refueling outage.
Calvert Cliffs	March 1993	Comp Drop	A two-block of the auxiliary hoist on the turbine building crane resulted in the cable breaking. The hook and block assembly fell approximately 12 meters (40 feet) hitting a section of reheat cross-over piping , a gang box, and then landed on the turbine deck, damaging the grating and concrete.
Fort Calhoun	March 1993	Load Drop	A mobile crane tipped over during a lift of ice deflectors. The crane fell onto and toppled a security camera tower.
Shoreham	April 1993	Load Drop	A refueling jib crane weighing approximately 4.5 metric tons (5 tons) fell from the polar crane auxiliary hook to the refueling floor when the nonredundant lifting eye broke. To balance the load the licensee attached a plasma arc welding machine to the lift. Minor injuries were received by a worker.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Sequoyah 1	June 1993	Load Drop	During fuel loading activities using the manipulator crane, an assembly was released prematurely, tilted over and came to rest against the south core baffle plate leaning at an angle of approximately 18 degrees from vertical. A phase A isolation, auxiliary building insolation, and containment ventilation isolation were manually initiated in accordance with procedures. No damage was done.
Vermont Yankee	Sept. 1993	Load Drop	When removing a fuel assembly from the reactor core, the assembly became detached from the grapple. The fuel assembly fell back into its original location in the core. Proper grapple engagement was not verified prior to movement.
Peach Bottom 2	Sept. 1993	Load Drop	An empty irradiated component shipping liner weighing 386 kg (850 pounds) was suspended from an auxiliary hook of the reactor building crane via an adapter about 2.1 meters (7 feet) below the surface of the spent fuel pool. It dropped approximately 6 meters (20 feet) into the cask storage area. The adapter hook was equipped with a safety latch designed to prevent the load from slipping off the hook. The safety latch had been taped back prior to being attached to the liner sling to facilitate removal of the hook from the sling.
Arkansas Nuclear 1	Sept. 1993	Load Slip	During the lift of a reactor vessel head, the polar crane's main hoist vertical motion was stopped and the head was trolleyed horizontally in the refueling canal. When the lift was resumed, the main hoist motor could not reestablish vertical motion. Subsequent attempts were made to reestablish vertical lift; but during each attempt, the head lowered slowly instead of rising.
Susquehanna 1	Oct. 1993	Load Slip	While lowering a fuel assembly into the core, an unexpected drop of one of the sections of the fuel handling mast occurred (25 to 38 centimeters [10 to 15 inches]). It was determined that the mast was damaged earlier during a collision.
Fermi 2	May 1994	Load Drop	A mobile crane weighing approximately 32 metric tons (35 tons) tipped over on its side when lifting a steel resin liner (weighing approximately 8.6 metric tons [9.5 tons]) to a transport truck. The boom struck the liner as the crane tipped over, partially crushing the top and bottom of the liner.
Hatch 1	Dec. 1994	Load Drop	Seven shroud bolts were being lifted from the spent fuel pool. When the bolts were about 31 centimeters (one foot) above the water, the rigging failed, and the bolt fell back into the pool, puncturing the stainless steel liner. Water from the pool drained into the area between the liner and the outer concrete wall causing the water level to drop about 7.6 centimeters (3 inches). The rigging was not correctly fabricated.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Indian Point 2	May 1996	Load Drop	A metal transportation container weighing approximately 2.3 metric tons (2.5 tons) was dropped when slings were set at too acute an angle, and the slings slipped off the crane hook.
Susquehanna	April 1997	Load Drop	While transporting a toolbox weighing approximately 1.8 metric tons (2 tons) using an auxiliary hoist on the reactor building crane, a nylon sling separated. One end of the box dropped approximately 1.8 meters (eight feet) striking the edge of a stored Unit 2 cavity shield plug. Routine testing of slings was found to be a weakness.
Waterford	April 1997	Load Drop	A new fuel assembly was dropped during fuel movements in the spent fuel pool. The cause was unknown.
Byron	Dec. 1997	Load Slip and Drop	During the lift of a steam generator replacement runway section weighing approximately 26 metric tons (28.5 tons), it slipped about 4.6 meters (15 feet) and came to an abrupt stop which caused the nylon rigging straps to fail. The runway section fell approximately 18 meters (60 feet) to the ground. Operator error was the most likely cause of the slip and drop.
Palo Verde 1	Feb. 1998	Load Drop	New fuel receipt inspection activities were being conducted in the Unit 1 fuel building. The shipping container had been unbolted and a lifting rig attached. The entire container was accidently lifted approximately 5 centimeters (2 inches) above the platform instead of just the lid. When this condition was realized, the decision was made to lower the container, when the lid separated and the fuel was dropped to the floor. No damage was done to the new fuel.
Davis-Besse	April 1998	Comp Drop	A wire support cable for the polar crane control pendant broke and caused a pendant with cabling (weighing a few hundred kilograms (several hundred pounds) to fall about 43 meters (140 feet), nearly missing personnel. No cause was given.
Davis-Besse	April 1998	BTH Drop	A jib arm on the polar crane trolley hit a winch cable supporting a ball and hook rigging device. The (rigging) device fell approximately 61 meters (200 feet) into the shallow end of the refueling canal missing personnel by 0.9 meter (3 feet).
Davis-Besse	April 1998	Load Drop	A portable transformer fell from a load being lifted.
Davis-Besse	April 1998	Load Drop	Rigging from eddy current equipment came loose, dropping the equipment 4.6 to 6.1 meters (15 to 20 feet).
Duane Arnold	May 1998	Load Drop	A main generator exciter coupling weighing 159 to 227 kgs (350 to 500 pounds) was dropped approximately 1.5 meters (5 feet) onto the turbine floor due to improper rigging.

Table 2: Reported crane events involving a load drop or a load slip (Continued)

Plant	Event Date	Event Type	Event Description
Grand Gulf	May 1998	Load Slip	A core shroud tool ring became dislodged from the strong back being used to lift the ring during a planned heavy lift to remove the ring from the reactor vessel. The ring became dislodged when operations personnel changed a system alignment so that a large volume of air rose from the reactor core. When the volume of air struck the ring and lifting rig, they shook violently, resulting in two adjacent suspension points becoming dislodged (there were four total suspension points). The ring was bearing against the top of the drywell flange, the drywell manway covers, and the drywell head studs. Review and evaluation of the lifting rig and photographs provided no information as to why the rig failed.
South Texas Project	Sept. 1998	Load Drop	A trailer used for snubber inspection was dropped from an elevation of about 30 centimeters (1 foot) on the Unit 2 Fuel Building truck bay. A leather glove used as a softener was insufficient to keep the sling from severing.
Trojan	April 1999	Load Drop	A mobile crane tipped over when lifting light poles because the outriggers were not extended.
Comanche Peak 1	Oct. 1999	Load Slip	During the removal of reactor coolant pump motor 1-03 weighing approximately 20 metric tons (22 tons), the electric hoist/chain fall failed. A hoist having a capacity of 41 metric tons (45 tons) was attached to the polar crane. When the hoist failed, the reactor coolant pump motor dropped approximately 4.6 to 6.1 meters (15 to 20 feet) in an unplanned descent before the hoist chain caught and prevented the motor from striking any plant structures or components. The hoist failed due to fatigue cracking of the spindle unit gear teeth. During testing prior to its use, the hoist malfunctioned. After several attempts at performing the test, the hoist began to function properly and the job proceeded. Improper assembly of the hoist following an overall was considered the root cause of failure.
Millstone 1	Oct. 1999	Load Slip	A new fuel bundle lifted by an auxiliary hoist on the reactor building crane continued to drift downward past its stop point until it came in contact with the refueling floor. No damage was done.
Crystal River	Nov. 1999	Load Slip	While lifting a reactor plenum with the polar crane, the lifting device became cocked because it was not attached properly.
Sequoyah	Dec. 1999	Load Drop	A mobile crane tipped over while moving a cell cap to a compartment at the low level waste facility.
Oyster Creek	Aug. 2000	Load Drop	Two new fuel assemblies fell from their metal container onto the refuel floor of the reactor building because of rigging issues. One worker received a glancing blow when the fuel fell.

Plant	Event Date	Event Type	Event Description
Haddam Neck	Oct. 2000	Load Drop	A radwaste canister filter dropped onto the spent fuel racks caused by rigging issues. No damage occurred.
Crystal River	March 2001	Load Slip	The spent fuel pool fuel handling hoist moves down on its own, caused by main hoist switch malfunctions.
North Anna 1	Mar. 2001	Load Drop	The top nozzle and fuel assembly weighing less than 0.9 metric ton (1 ton) separated which allowed the fuel assembly to drop approximately 3.7 meters (12 feet). The break was caused by IGSCC.
San Onofre 3	May 2001	Load Drop	A mobile crane weighing approximately 34 metric tons (37.5 tons) was dropped approximately 12 meters (40 feet) to the Unit 3 turbine bay floor when Kevlar slings failed. The mobile crane was severely damaged. The sling failure was caused by using inadequate rigging softener material.
Turkey Point 4	June 2001	Load Drop	A Kevlar sling separated causing the drop of a Link-Belt mobile crane weighing approximately 34 metric tons (37.5 tons) to the Number 4 turbine building laydown area. The total drop was of approximately 20 centimeters (8 inches). The sling separated because it was not properly protected at sharp corners. The mobile crane was inspected and found not to be damaged.
Peach Bottom 2	Sept. 2002	Load Drop	Unit 2 B recirculation pump motor dropped approximately 25 centimeters (10 inches) to its motor stand because the hoist chain broke. The hoist was not tested to 125 percent of load prior to use. The hoist chain was not the correct material. Plant conditions were not established prior to the load movement (e.g., a subsystem of the RHR shutdown cooling was not operable in case of a load drop).

2.3.11 Crane Types Resulting in Deaths or Injuries

From an occupational safety standpoint, crane events leading to death or injury were reviewed. For the period 1968 through 2002 there were 10 events that resulted in a death, and 16 events that resulted in an injury. Figure 11, "Crane types resulting in deaths or injuries," shows the number of events that led to either a death, an injury, or both a death and an injury. In reviewing deaths and injuries caused by crane operation, each event was sorted by crane type. Crane types were put into eight different categories.

Auxiliary building: Larger capacity crane in the auxiliary building.

Mobile: Movable crane having various arrangements of fixed or telescoping booms or jibs. Generally used during both construction and maintenance activities.

Other: Any of several cranes not fitting into other categories (i.e., turbine building, fuel storage cask, fuel building, radwaste building, or other cranes not specifically identified by type.

Polar: Large capacity overhead crane that operates on a circular runway, normally located inside of the containment building.

Reactor Building: Large capacity overhead crane operating on a parallel runway.

Refueling/Manipulator: Low capacity bridge crane used to defueling and refueling operations.

Spent Fuel Pool: Various types of bridge cranes. Used for moving spent fuel from one location to another.

Tower: Consists of a vertical tower and either a fixed or movable jib. Generally used during initial construction.

As shown by Figure 11, there were three crane events that involved an RB or Polar crane out of the total of 27 events that resulted in either a death or injury. Most deaths and injuries occurred while using cranes that typically do not move loads that impact safety-related equipment (i.e., tower, mobile, or other categories). These types of cranes have typically not been as well controlled and maintained in the past as are polar, reactor building, or spent fuel pool cranes. Mobile cranes and Other cranes represent the highest percentage of both deaths and injuries. See Section 2.3.12 for a discussion of events that led to a death.

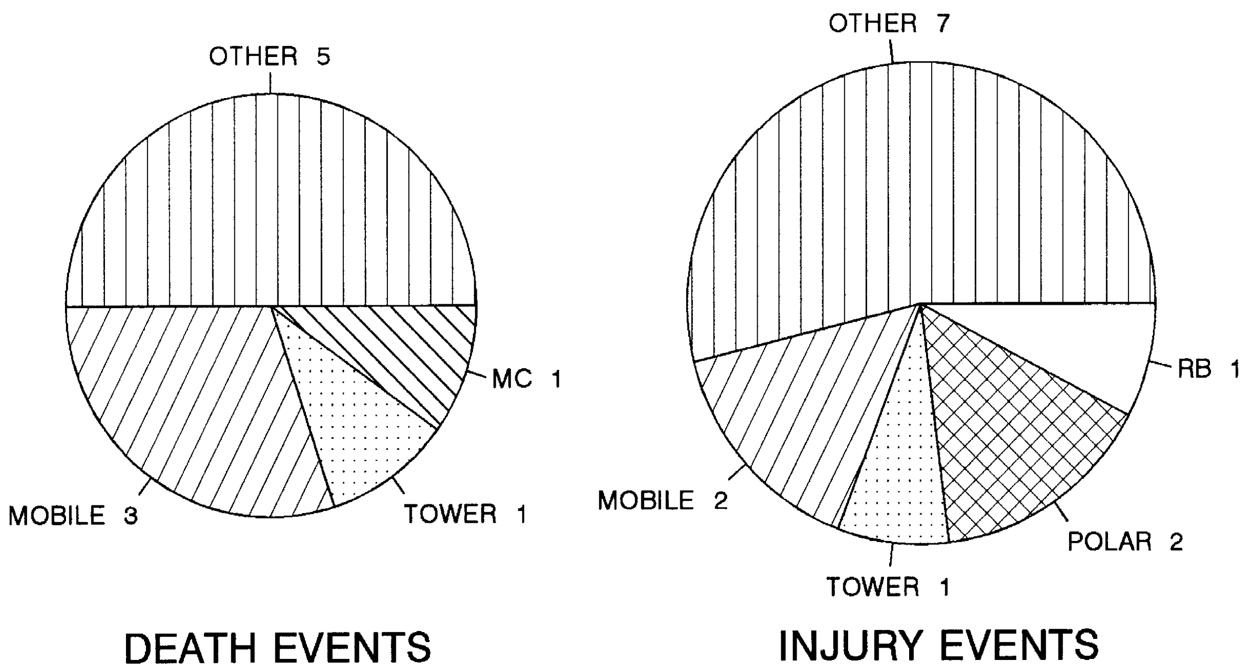

Figure 11: Crane types resulting in deaths or injuries

2.3.12 Description and Distribution of Crane Related Deaths

Figure 12, "Distribution of crane related deaths," shows the number of crane related events leading to at least one death. In some instances, one crane event resulted in multiple deaths. There have been 10 reported crane events that have led to deaths in the nuclear industry for the period 1968 through 2002. The highest concentration of crane related deaths at nuclear power plants occurred during the first decade (1969 to 1980). For Decade 1, six of eight events that led to a death occurred at facilities still under construction. The last death in a crane related accident in the U.S. nuclear industry was Decade 2 (1985). Table 3, "Reported crane events resulting in deaths" provides information for each of the reported crane events that involved a death. Figure 12 also shows the cumulative number of nuclear power plants that had an operating license during the period from 1968 through 2002.

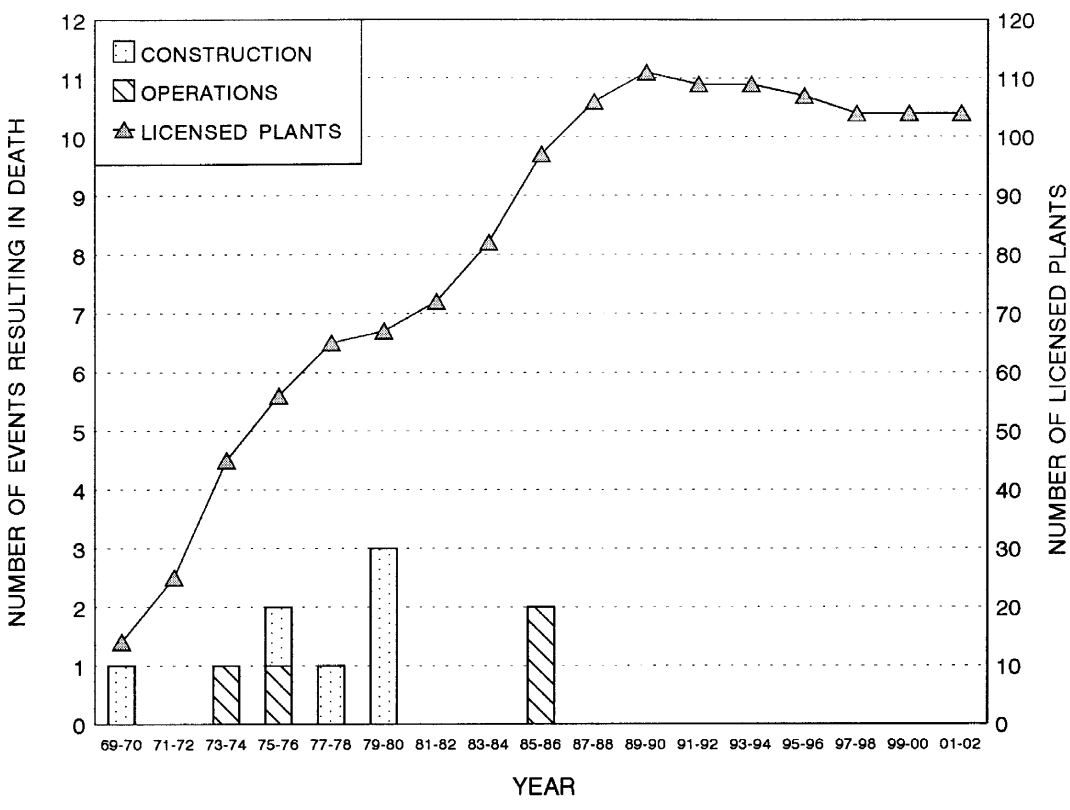

Figure 12: Distribution of crane related deaths

Table 3: Reported crane events resulting in deaths

Plant	Event Date	Event Description
Turkey Point 4	March 1970	The main generator stator for Unit 4, which was to be installed in Unit 3, dropped 31 to 61 centimeters (one to two feet) when two vertical crane support cables snapped during a lifting operation. The support columns for the portable crane also collapsed. One section of the support columns struck and killed an engineer. Other falling sections injured two other personnel. Some turbine piping was damaged but no nuclear components were affected.
Haddam Neck	Dec. 1973	A worker died following a 3 meters (10 feet) fall from an overhead yard crane.
Peach Bottom 2,3	May 1976	A contractor employee fell 15 meters (50 feet) to his death while riding a crane hook in the radwaste building.
Comanche Peak 1,2	May 1976	Failure of a portable crane boom resulted in the deaths of two construction employees when the crane became unbalanced and the boom and a occupied personnel bucket fell to the turbine mat area.
Nine Mile Point 2	Feb. 1978	Two workers were killed when a section of installed reinforcing bars collapsed when struck by a bundle of reinforcing bars being handled by a crane.
Perry 1,2	Oct. 1979	A worker was killed when he touched a crane which was in contact with a high voltage overhead line.
Marble Hill 1,2	Feb. 1980	A worker was killed when a mobile crane got stuck in the mud and tipped over while the operator was raising the load to try to free the crane.
Byron 2	Aug. 1980	A worker was killed when he was caught between a crane counterweight and the engine housing.
McGuire 2	Feb. 1985	An equipment operator was killed when he attempted to step onto a moving manipulator crane and fell back and lodged his head between the crane and an electrical lighting panel.
Brown’s Ferry 2	March 1985	A maintenance worker was killed and three others were injured when they were struck by a falling crane hook inside the unit 2 turbine building. The accident occurred when the overhead crane cable parted. A crane hook having a capacity of 23 metric tons (25 tons) dropped through the roof of a temporary building where the maintenance workers were located.

2.3.13 Description and Distribution of Crane Related Injuries

Figure 13, “Distribution of crane related injuries,” shows the number of crane related events leading to at least one injury. In some instances, one crane event resulted in multiple injuries. There have been 16 reported crane events that have led to injuries in the nuclear industry for the period 1968 through 2002.

Decade 1. Seven injuries, six of which occurred at plants under construction.

Decade 2. Three injuries, two of which occurred at plants under construction.

Decade 3. Six injuries, all occurring at operating plants. When comparing Decade 2 crane injuries with Decade 3 injuries, a 100 percent increase in the number of injuries occurred with a 9 percent increase in the number of operating power plants.

Figure 13 also shows the cumulative number of nuclear power plants that had an operating license during the period from 1968 through 2002. When considering the number of operating facilities during each of the three decades, the injury rate has significantly decreased.

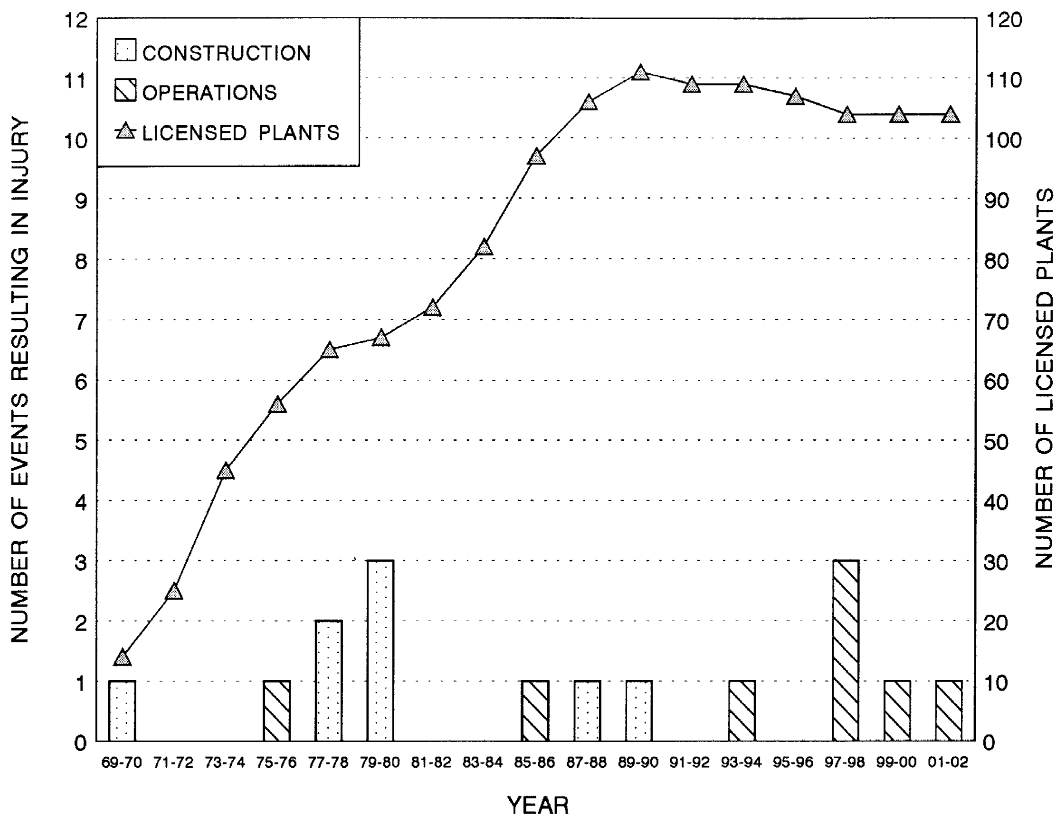

Figure 13: Distribution of crane related injuries

2.3.14 Distribution of Fuel Assembly Drops or Fuel Handling Damage

There have been 30 crane events involving either a fuel assembly drop or damage to a fuel assembly caused by handling. There was a steady increase in the number of operating units to over 100 during the period of the survey that was also accompanied by a steady overall improvement in fuel handling performance. Figure 14, “Distribution of fuel assembly load drops or fuel handling damage events,” shows the event distribution in two-year increments. Fuel assembly events resulting in a drop or fuel damage are listed in Table 4: “Fuel assembly load drop or fuel handling damage events.” Many of the fuel assembly issues have been captured in NRC generic communications listed in Appendix G. From a risk perspective, none of the 30 fuel assembly drop or fuel handling events resulted in radiation exposure or risk to personnel. A comparison of the three decades presents the following observations:

Decade 1. There were 11 events, two of which involved fuel handling damage. During this decade, the average number of operating units was 45.

Decade 2. There were 12 events, one involved fuel handling damage. During Decade 2, there was a positive trend. The number of events increased by 9 percent, however the average number of operating units increased by 113 percent when compared to Decade 1.

Decade 3. There were 7 events, one involved fuel handling damage. During Decade 3, another positive trend occurred; the number of events decreased by 33 percent, concurrent with an increase in the number of operating units by 9 percent when compared to Decade 2.

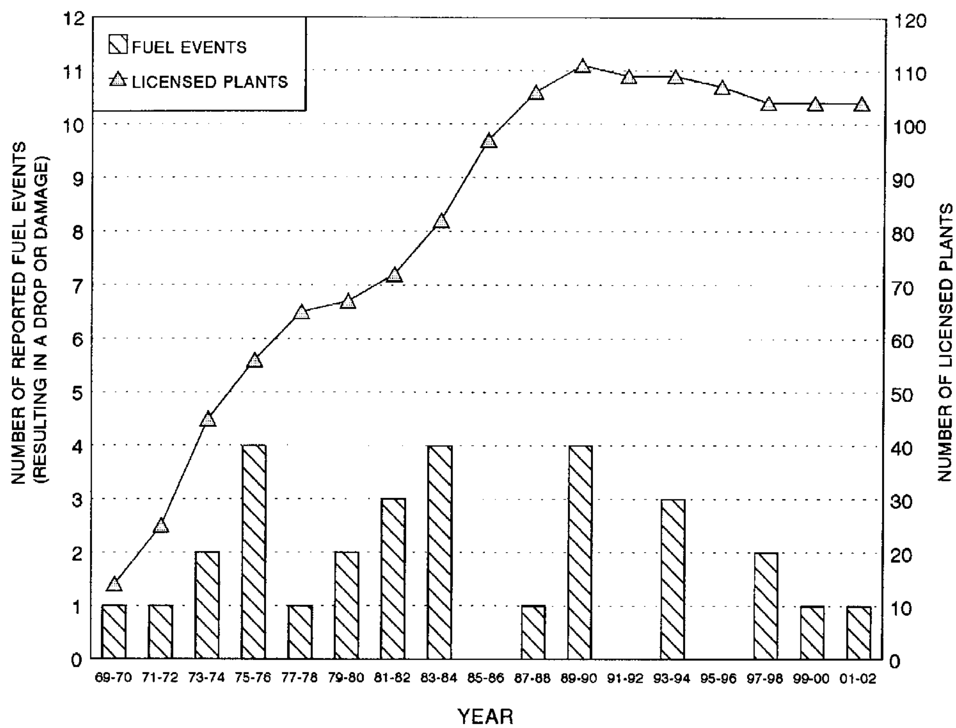

Figure 14: Distribution of fuel assembly load drops or fuel handling damage events

Table 4: Fuel assembly load drop or fuel handling damage events

Plant	Event Date	Event Type	Event Description
Yankee Rowe	Aug. 1969	Fuel Damage	During refueling, a fuel bundle was damaged. The assembly retainer band on the carriage had been crushed against the fuel assembly lower nozzle. In addition, a welded joint on the upender was cracked. The event was caused by the crane operator.
Fermi 1	Oct. 1972	Load Drop	While transferring fuel from an auxiliary fuel storage facility to the Fuel and Repair Building, a crane operator inadvertently actuated the “raise” instead of the “lower” control, causing the 1/4" bolt in the shackle holding the subassembly to fail. As a result, the subassembly fell 8 meters (27 feet) into the transfer tank.
Pilgrim	Jan. 1974	Load Drop	An irradiated fuel assembly became detached from the grapple and fell in the spent fuel pool.
Millstone 1	Sept. 1974	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool.
Duane Arnold	June 1975	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool.
Humbolt Bay	June 1975	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool.
Brunswick 2	March 1976	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool. The assembly fell before it was fully inserted into its rack.
Brunswick 2	March 1976	Load Drop	A fuel assembly became detached from the grapple and fell in the spent fuel pool. The assembly fell to a horizontal position across the top of the spent fuel pool storage racks.
Peach Bottom 3	Jan. 1977	Load Drop	A fuel assembly was inadvertently released from the grapple and fell across the core. The cause was attributed to operation of the grapple open switch on operator error.
Crystal River	June 1978	Load Drop	A missile shield crane hook failed, dropping its test weight on a fuel assembly causing minor damage. The crane hook was plant fabricated.
Salem 1	May 1979	Fuel Damage	A total of 31 fuel assemblies that were removed from the core had suffered some grid damage due to load movements.
Pilgrim	Dec. 1979	Load Drop	A new fuel assembly was being moved to the spent fuel pool using the reactor building crane, when the assembly struck the top edge of the high density fuel racks and the latching device on the auxiliary hook failed to retain the fuel. The assembly fell, striking the lifting bails on four spent fuel elements, then coming to rest on the top of the fuel racks.

Table 4: Fuel assembly load drop or fuel handling damage events (Continued)

Plant	Event Date	Event Type	Event Description
Prairie Island 1	Jan. 1981	Load Drop	The top nozzle and fuel assembly separated, which led to the drop of an assembly weighing less than 0.9 metric ton (1 ton). The break was caused by IGSCC.
Cook 1	June 1981	Fuel Damage	A fuel bundle was damaged while it was being transferred in the refueling cavity using the manipulator crane, when the lower end of the assembly struck a ledge on the refueling cavity floor just outside the reactor vessel area. One rod was dislodged and fell from the assembly onto the refueling cavity floor. No radiation was released.
Cook 1	April 1982	Load Drop	The upender device had not been raised to the vertical position before the fuel assembly was lowered. This resulted in the fuel assembly becoming cocked and lodged in the manipulator mast. Minor deformation marks and scratches were noticed on a few rods. No radiation was released.
Turkey Point 4	April 1983	Load Drop	A fuel assembly was being inserted into the core. It was not aligned properly, and fell over so that is leaned at a 35 degree angle against two other fuel assemblies.
Turkey Point 4	April 1983	Load Drop	During a fuel bundle lift from its storage rack, the limit switches failed to stop upward movement. The hoist two blocked, parting the hoist cable and causing the assembly to drop back into its rack.
Hatch 1	Oct. 1984	Load Drop	A possible inadvertent actuation of the fuel grapple hook position switch resulted in dropping a spent fuel bundle about 3.7 meters (12 feet) into its storage rack cell, slightly deforming and scratching the bundle and rack. No radiation release occurred.
Millstone 2	Nov. 1984	Load Drop	A spent fuel pin was dropped while performing eddy current testing. The cause was attributed to inadequate gripping force on the pin.
Grand Gulf	July 1987	Load Drop	A container of two new fuel bundles fell off a transfer cart to the turbine deck because of crane operator and rigging issues. Both fuel bundles had minor damage and were not used.

Table 4: Fuel assembly load drop or fuel handling damage events (Continued)

Plant	Event Date	Event Type	Event Description
Quad Cities 1	Sept. 1989	Load Drop	During the transfer of new fuel from the new fuel storage vault to the fuel pool, a fuel assembly was released from the refueling grapple and fell upon the spent fuel racks. The grapple control switch was left in the "release" position when it was decided to lift the fuel to reposition it. The fuel was released, falling to the rack. The dropped fuel assembly, including the irradiated fuel it fell on, were visually examined in place from the bridge and the floor for signs of fuel damage. No damage was observed. Although no apparent damage resulted the fuel, 12 of the 32 potentially impacted fuel assemblies were discharged instead of reloaded for use in the next fuel cycle. The dropped fuel bundle was to be returned to GE.
North Anna 1	Jan. 1990	Load Drop	While the fuel building ventilation system was not aligned to discharge through the auxiliary building HEPA filter and charcoal absorber assembly, one fuel rod inadvertently slipped from the fuel rod handling tool due to a mechanical failure of the gripper mechanism, and dropped into its proper storage location in an uncontrolled manner. The height of the drop was not recorded, but no damage was recorded.
Byron 2	Sept. 1990	Load Drop	A fuel assembly slipped out of the basket and dropped to the top of an empty fuel rack.
Indian Point 3	Oct. 1990	Load Drop	While lifting the upper core support structure weighing approximately 54 metric tons (60 tons), two fuel assemblies were found to be attached. One of the assemblies dropped into a retrieval basket when the brakes on the overhead crane were applied. A guide pin on each assembly was bent. The guide pins were most likely damaged during the previous refueling outage.
Sequoyah 1	June 1993	Load Drop	During fuel loading activities using the manipulator crane, an assembly was released prematurely, tilted over and came to rest against the south core baffle plate leaning at an angle of approximately 18 degrees from vertical. A phase "A" isolation, auxiliary building insolation, and containment ventilation isolation were manually initiated in accordance with procedures. No damage was done.
Vermont Yankee	Sept. 1993	Load Drop	When removing a fuel assembly from the reactor core, the assembly became detached from the grapple. The fuel assembly fell back into its original location in the core. Proper grapple engagement was not verified prior to movement.
Quad Cities 1	June 1994	Fuel Damage	While lowering a fuel assembly, it was improperly inserted, causing damage to the assembly and the fuel handling crane.

Table 4: Fuel assembly load drop or fuel handling damage events (Continued)

Plant	Event Date	Event Type	Event Description
Waterford	April 1997	Load Drop	A new fuel assembly was dropped during fuel movements in the spent fuel pool. The cause was unknown.
Palo Verde 1	Feb. 1998	Load Drop	New fuel receipt inspection activities were being conducted in the Unit 1 fuel building. The shipping container had been unbolted and a lifting rig attached. The entire container was accidently lifted approximately 5 centimeters (2 inches) above the platform instead of just the lid. When this condition was realized, the decision was made to lower the container, when the lid separated and the fuel was dropped to the floor. No damage was done to the new fuel.
Oyster Creek	Aug. 2000	Load Drop	Two new fuel assemblies fell from their metal container onto the refuel floor of the reactor building because of rigging issues. One worker received a glancing blow when the fuel fell.
North Anna 1	Mar. 2001	Load Drop	The top nozzle and fuel assembly weighing less than 0.9 metric ton (1 ton) separated which allowed the fuel assembly to drop approximately 3.7 meters (12 feet). The break was caused by IGSCC.

2.3.15 Distribution of Events Involving Mobile Cranes

There have been 38 recorded events involving mobile crane operation from 1968 though 2002. Many of these resulted in tip overs, load drops, and equipment damage. Several mobile crane events have resulted in a loss or partial loss of power to various electrical lines servicing plant equipment. Loss of power events are described in Section 2.3.16. Figure 15, "Distribution of events involving mobile cranes," shows an overall negative trend in the number of events. A comparison of the three decades presents the following observations:

Decade 1. There were six events of which five occurred at plants under construction. During this decade, the average number of operating units was 45.

Decade 2. There were 17 events of which four occurred at plants under construction. During Decade 2, a negative event trend occurred. The number of events increased by approximately 183 percent, while the average number of operating units increased by 113 percent when compared to Decade 1.

Decade 3. There were 15 events at operating facilities. During Decade 3, event rate remains essentially constant with Decade 2. The number of events decreased by 12 percent, concurrent with an increase in the number of operating units by 9 percent when compared to Decade 2.

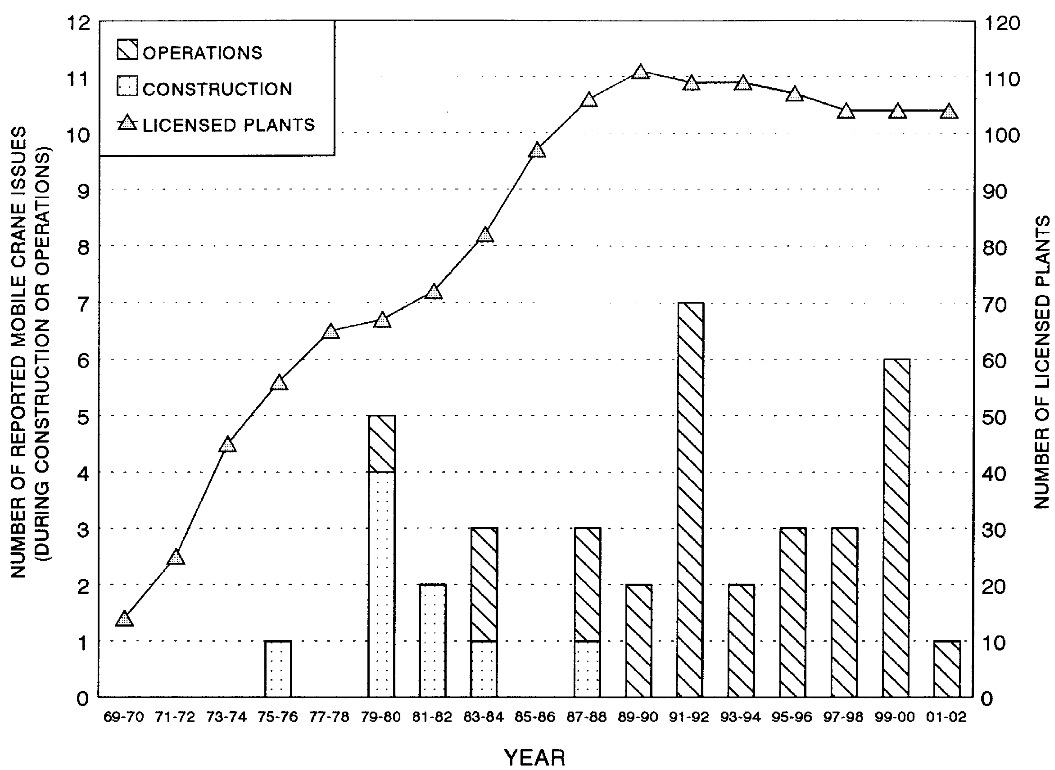

Figure 15: Distribution of events involving mobile cranes

2.3.16 Loss of Power Events Involving Crane Operation

There have been 10 loss of power events caused by crane operation during the time period of 1968 through 2002, nine of which were caused by mobile cranes. Of the 38 crane events involving mobile cranes shown in figure 15, nine have resulted in a loss or partial loss of power. Most nuclear plants show their greatest risk in the area of loss of offsite power, consequently, crane events that produce this kind of challenge to a plant is of increased interest from a risk perspective. Table 5, "Crane events resulting in a loss or partial loss of power," provides a brief description of loss of power crane events. Of the nine crane events described in Table 5, two licensees had Augmented Inspection Team (AIT) inspections (Palo Verde and Diablo Canyon). However, none of the 10 crane events met the minimum risk threshold requirements to be classified as an ASP event. Figure 16, "Crane events resulting in a loss of power," shows an overall negative trend in crane events resulting in a loss of power, but an improving trend within the last few years. A comparison of the three decades presents the following observations:

Decade 1. There were no crane events resulting in a loss of power while the average number of operating units was 45.

Decade 2. There were seven events that resulted in a loss of power. During Decade 2, the average number of operating units increased by 113 percent when compared to Decade 1.

Decade 3. There were three events that resulted in a loss of power. During Decade 3, the number of events decreased by 43 percent, although the average number of operating units increased by approximately 9 percent compared to Decade 2.

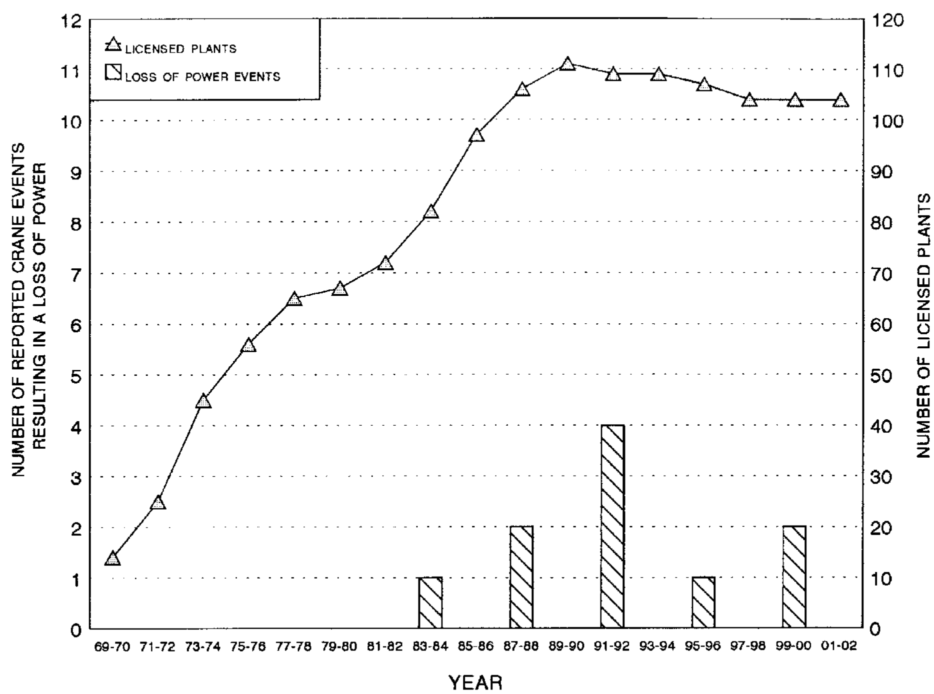

Figure 16: Crane events resulting in a loss of power

Table 5: Crane events resulting in a loss or partial loss of power

Plant	Event Date	Description
Three Mile Island 2	Aug. 1983	A mobile crane at TMI-2 made contact with a power line (230 kV) which was a source of offsite power for both units. It resulted in the loss of one of two trains of safety related electrical distribution busses in both units. The crane was not loaded at the time of the event.
Crystal River	July 1987	A small crane hook collided with a breaker cubicle, shorting two phases and causing an under voltage. This caused the "A" engineered safeguards train to be inoperable, while the "B" train was out of service.
Peach Bottom 2	Aug. 1987	While Unit 2 had been shutdown for five months, a mobile crane contacted an energized 220 kV line resulting in tripping of the Unit 2 startup source. Both Units 2 and 3 were effected. Unit 3 “C” RHR was restored within 10 minutes. Unit 2 “C” RWCU pump was restored within 37 minutes, and RHR was returned to service within 4 hours. (LER 277-87-016)
Diablo Canyon 1	March 1991	A mobile crane shorted the "A" phase main transformer to ground (500 kV line). The remaining electrical line, a 230 kV line was out for maintenance. The 230 kV line was restored in about 5 hours.
Palo Verde 3	Nov. 1991	While Unit 3 was in hot standby, a mobile crane contacted a 13.8 kV overhead line causing a partial loss of offsite power. The crane was not grounded, was not level, the friction brake was not set, and the crane was left unattended when its boom rotated into the power line. (LER 530-91-010-01, also an augmented inspection team (AIT) inspection was performed)
Fermi 2	Dec. 1991	While in cold shutdown, a mobile crane contacted an energized 120 kV overhead electrical line twice. The circuit opened and closed momentarily for each contact, but did not cause a loss of offsite power. After the first contact, the driver of the mobile crane backed into the line a second time. (No LER was written)
Nine Mile Point 2	Sept. 1992	While Unit 2 was at 100 percent power, a mobile crane boom got too close to one of two 115 kV lines, tripping the line and causing a partial loss of offsite power. Division I and II EDGs ran loaded for approximately 4 hours each. The 115 kV line was restored within approximately 3 hours. (LER 410-92-020)
Indian Point 3	March 1995	While Unit 3 was in cold shutdown, a mobile crane in the Indian Point 2 owner controlled area shorted the “C” phase of the 138 kV feeder to ground causing a loss of offsite power. The event occurred while the mobile crane was loading material into a flatbed truck. Emergency power was provided by two EDGs. (LER 286-95-004)
Dresden	May 1999	While lifting valves and pipe fittings from a laydown area, a mobile crane boom came too close to a 34 kV line. The line tripped and then reclosed. No injuries resulted.
Braidwood	July 2000	The loss of a 34 kV line to the river screen house, and subsequent tripping of the CW pumps was caused by a loaded mobile crane. There was no spotter for the crane operator. No damage was done to the crane, or the operator. The 34 kV line was damaged but not severed.

2.3.17 Distribution of Below-the-Hook Crane Events

For the period 1968 through 2002, there have been 47 below-the-hook events that have been reported. A below-the-hook event is classified as an event where rigging or handling errors resulted in an event. Figure 17, "Below-the-hook crane events," shows a significant overall increase in the number of events. Figure 17 also categorizes below-the-hook events into "administrative," "equipment damage," "load slip," and "load drop." Many of these events have resulted in load drops and damaged equipment. A comparison of the three decades presents the following observations:

Decade 1. There were four below-the-hook events. Three events resulted in equipment damage, and one event resulted in a load drop. During Decade 1, the average number of operating units was 45.

Decade 2. There were 10 below-the-hook events. Six involved a load drop, three involved equipment damage, and one involved a load slip. During Decade 2, the number of events increased by 150 percent, concurrent with an increase in the number of operating units by 113 percent when compared to Decade 1.

Decade 3. There were 33 below-the-hook events, of which 17 involved load drops, 10 involved equipment damage, four involved administrative issues, and two involved load slips. During Decade 3, the number of events increased by 230 percent, concurrent with an increase in the number of operating units by 9 percent when compare to Decade 2.

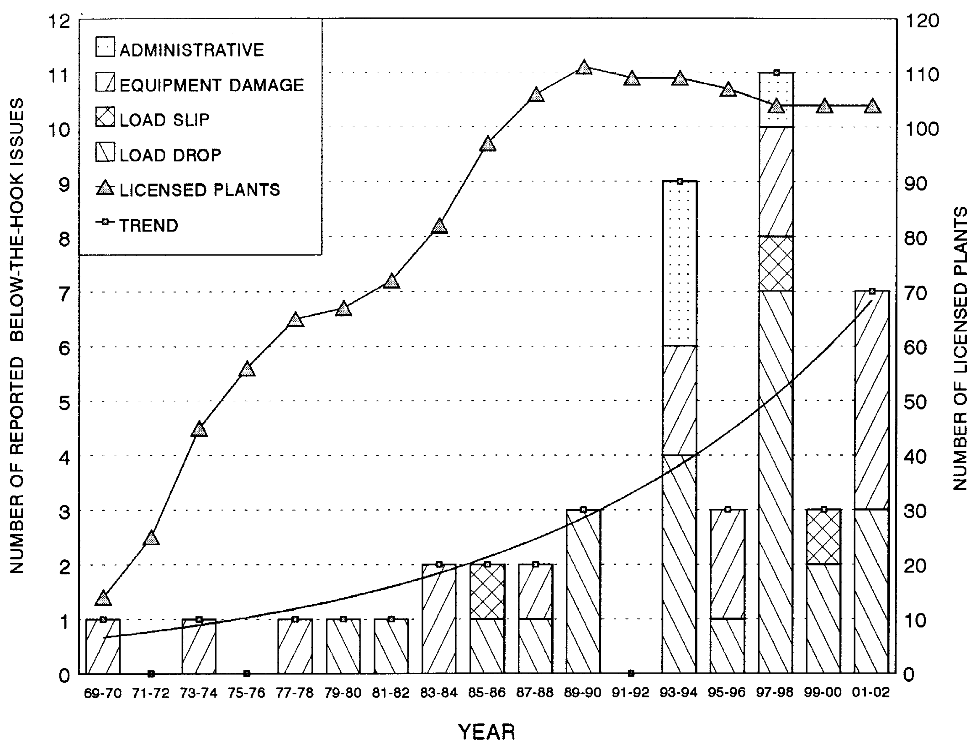

Figure 17: Below-the-hook crane events

Table 6: Below-the hook crane events

Plant	Event Date	Description
Yankee Rowe	Aug. 1969	During refueling, a fuel bundle was damaged. The assembly retainer band on the carriage had been crushed against the fuel assembly lower nozzle. In addition, a welded joint on the upender was cracked. The event was caused by operator error.
Vermont Yankee	March 1973	A grapple came lose from the jib crane hoist cable.
Callaway	Aug. 1978	Two workers were injured when a support girder for the polar crane rail fell on them. Rigging caught on the girder, causing it to fall.
Shearon Harris	May 1980	A crane inside containment lifted a lifting tackle approximately 43 meters (140 feet) when the binding broke. The lifting tackle fell, landed first on scaffolding, and then onto eight workers. Various injuries were received. The cause is unknown.
Prairie Island 1	Jan. 1981	The top nozzle and fuel assembly separated causing the assembly weighing less than 0.9 metric ton (1 ton) to drop . The failure was caused by IGSCC.
Perry	Sept. 1983	While attempting to remove the shroud head/separator weighing approximately 45 metric tons (50 tons) from the reactor pressure vessel, the strongback lifting device was broken, because the securing fasteners were not removed prior to the lift.
Rancho Seco	Feb. 1984	Slings used to lift the refueling rack failed due to excessive load, caused in part by a load cell that was set too high, and by improper rigging. The rack was safely lowered without dropping.
St. Lucie	Nov. 1985	While performing a lift of the upper guide structure weighing approximately 45 metric tons (50 tons), a bolt used to help secure a rigging device failed because it was improperly threaded. The upper guide structure tilted approximately 15 centimeters (6 inches) when the bolt failed. No damage was done.
Three Mile Island 2	Dec. 1985	A defueling canister and support sleeve weighing approximately 1 metric ton (1.1 tons) fell into the reactor vessel when is was dislodged from the positioning system using a jib crane. In addition, the jib crane was rated at 0.9 metric ton (1 ton), while the load was 1 metric ton (1.1 tons).
Grand Gulf	July 1987	A container of two new fuel bundles fell off a transfer cart to the turbine deck because of crane operator and rigging issues. Both fuel bundles had minor damage and were not used.
Crystal River	July 1987	A small crane hook collided with a breaker cubicle, shorting two phases and causing an under voltage. This caused the "A" engineered safeguards train to be inoperable, while the "B" train was out of service.
Quad Cities 1	Sept. 1989	While moving a new fuel bundle, it became detached from its grapple and fell onto fuel in the spent fuel pool.

Table 6: Below-the hook crane events (Continued)

Plant	Event Date	Description
Byron 2	Sept. 1990	A fuel assembly slipped out of the basket and dropped to the top of an empty fuel rack.
North Anna 1	Jan. 1990	Fuel rod slipped out of the handling tool and dropped into its storage location due to a gripper mechanism failure.
Shoreham	April 1993	A refueling jib crane weighing approximately 4.5 metric tons (5 tons) fell from the polar crane auxiliary hook to the refueling floor when the nonredundant lifting eye broke. To balance the load the licensee attached a plasma arc welding machine to the lifting device. Minor injuries were received by a worker.
Nine Mile Point 2	Nov. 1993	A blade guide was being moved from the core into the spent fuel pool when it was released from the grapple. The operator then moved the crane and noticed that the blade guide had never been released. The operator then tried to move a fuel assembly, and discovered that the mast was bent.
Peach Bottom	Sept. 1993	An empty component shipping liner weighing approximately 386 kgs (850 pounds) became disconnected because of rigging issues, and fell into the spent fuel pool cask storage area. No damage was done.
Oconee 3	Sept. 1993	An empty dry storage cask was placed in the Unit 3 spent fuel pool and was mispositioned on the cask pit stand. This resulted in the cask leaning to one side, which caused the lifting hook to partially slip off the cask trunnion during a lift attempt.
Shoreham	June 1993	Heavy loads were moved in the vicinity of the spent fuel pool with an incorrect lifting attachment.
Susquehanna 2	Oct. 1993	While transferring a double blade guide to the spent fuel pool, the blade guide hit the side of the reactor vessel because it was not raised high enough to clear the vessel. The following day, it was discovered that the mast was bent.
Sequoyah 1	June 1993	During fuel loading, a bundle was inappropriately unlatched, failed to insert, and tilted against the core baffle plate at an angle of approximately 18 degrees from vertical.
Hatch 1	Dec. 1994	Seven shroud bolts were being lifted from the spent fuel pool. When the bolts were about one foot above the water, the rigging failed, and the bolt fell back into the pool, puncturing the stainless steel liner. Water from the pool drained into the area between the liner and the outer concrete wall causing the water level to drop about 7.6 centimeters (three inches). The rigging was not correctly fabricated.

Table 6: Below-the hook crane events (Continued)

Plant	Event Date	Description
Waterford	Feb. 1994	An unknown object was found to be hanging from the fuel handling machine in the spent fuel pool. The object was determined to be a fuel rod encapsulation tube that stuck onto the fuel handling tool. The licensee could not determine the cause of the event.
Callaway	Oct. 1996	A sling failed as a load was lifted (reactor coolant pump motor). The lift was performed outside the power block.
Indian Point 2	May 1996	A metal transportation container weighing approximately 2.3 metric tons (2.5 tons) was dropped when slings were set at too acute an angle. The slings slipped off crane hook.
San Onofre	Dec. 1996	While lifting a section of the turbine hood, it became unbalanced and impacted a structural wall, damaging concrete.
Susquehanna 1	April 1997	While transporting a toolbox weighing approximately 1.8 metric tons (2 tons) using the reactor building crane auxiliary hoist, a sling parted, dropping the toolbox.
Catawba	Nov. 1997	The wrong softener material was used for rigging protection during movement of a spent fuel pool gate.
Byron 2	Dec. 1997	During the lift of a steam generator replacement runway section weighing approximately 26 metric tons (28.5 tons) located outside of containment, the runway section slipped about 4.6 meters (15 feet), came to an abrupt stop which caused the nylon rigging straps to fail. The runway section fell approximately 18 meters (60 feet) to the ground. Operator error was the most likely cause of the drop.
Palo Verde 1	Feb. 1998	During receipt of new fuel in the fuel building, a loaded container was dropped when the container lid separated from the bottom portion of the container.
Davis-Besse	April 1998	Rigging used to lift eddy current equipment came loose, resulting in a drop of the equipment 4.6 to 6.1 meters (15 to 20 feet).
Duane Arnold	May 1998	A main generator exciter coupling weighing from 159 to 227 kgs (350 to 500 pounds) was dropped approximately 1.5 meters (5 feet) onto the turbine floor due to improper rigging.
Grand Gulf	March 1998	A heavy load consisting of the core shroud inspection tool theta drive ring became partially disconnected from its strongback while being moved over irradiated fuel.
Grand Gulf	May 1998	A core shroud tool ring weighing 676 kgs (1490 pounds) became dislodged from its strongback during an accidental release of air through the reactor vessel.

Table 6: Below-the hook crane events (Continued)

Plant	Event Date	Description
South Texas Project	Sept. 1998	A trailer used for snubber inspection was dropped from an elevation of about 25 centimeters (1 foot) on the Unit 2 Fuel Building truck bay. The leather glove that was used as a softener was insufficient to keep the sling from severing.
Duane Arnold	May 1998	A turbine building crane load shifted because of rigging issues. A worker was injured when the load struck him.
Davis-Besse	April 1998	A portable transformer fell from a load that was being lifted.
Crystal River	Nov. 1999	While using the polar crane, the lifting device for the reactor plenum was not attached properly and the load became cocked.
Haddam Neck	Oct. 2000	A radwaste canister filter dropped onto spent fuel racks caused by rigging deficiencies. No damage occurred.
Oyster Creek	Aug. 2000	Two new fuel assemblies fell from their metal container onto the refuel floor of the reactor building because of rigging issues. One worker received a glancing blow when the fuel fell.
Byron 2	April 2001	While lifting a reactor stud bolt rack out of the reactor cavity, the stud rack caught on the handrail, pulling it loose. The handrail fell approximately 9.5 meters (31 feet) before striking a worker below. The worker was injured but equipment was not damaged.
Turkey Point 4	June 2001	A Link-Belt mobile crane weighing approximately 34 metric tons (37.5 tons) dropped approximately 20 centimeters (8 inches) to the Number 4 turbine building laydown area following the failure of a Kevlar sling. The sling separated because it was not properly protected at sharp corners. The mobile crane was inspected and found not to be damaged.
North Anna 1	March 2001	The top nozzle and fuel assembly separated, leading to the drop of an assembly weighing less than 0.9 metric ton (1 ton). The break was caused by IGSCC.
San Onofre 3	May 2001	A mobile crane weighing approximately 34 metric tons (37.5 tons) was dropped approximately 12 meters (40 feet) to the Unit 3 turbine bay floor when Kevlar slings failed. The mobile crane was severely damaged. The sling failure was caused by using inadequate rigging softener material.
Byron 1	March 2002	A liquid nitrogen bottle was damaged when it was hit by the auxiliary hook on the polar crane.
Point Beach	Dec. 2002	An auxiliary building crane pendant cable became entangled with the spent fuel pool bridge during preparations for cask loading. The spent fuel pool crane shook, and operators observed sparks on the auxiliary crane. Power was lost to the auxiliary crane.

Table 6: Below-the hook crane events (Continued)

Plant	Event Date	Description
Byron 1	March 2002	During movement of the Unit 1 reactor head stud rack weighing approximately 4 metric tons (4.5 tons), it contacted the maintenance hoist, damaging the 480 V hoist bus bar. The Kevlar sling was also discovered to be over loaded and had stretched.

2.3.18 Distribution of Crane Events by Plant

Figure 18, “Distribution of crane issues by facility, on a per unit basis,” shows the number of crane issues documented against each nuclear power plant facility, divided by the number of units (i.e. either units that received an operating license, or that were substantially completed) at that facility. Since there were many facilities that had units canceled, judgement was used in determining how many plants were “substantially” completed, but did not receive an operating license. Two nuclear facilities reported no crane events; Kewaunee and Watts Bar.

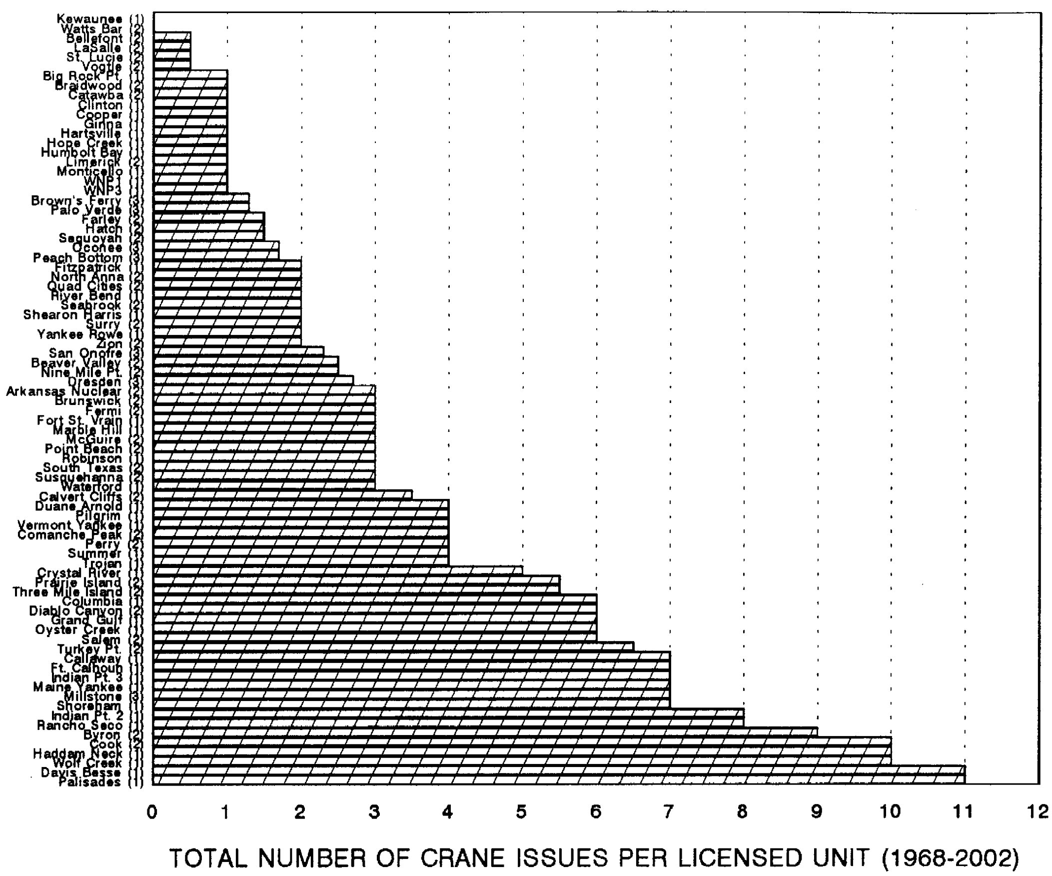

Figure 18: Distribution of crane issues by facility, on a per unit basis

3 LICENSEE CRANE OPERATING EXPERIENCE INVOLVING VERY HEAVY LOADS SINCE COMMERCIAL OPERATION

At an initial NRC meeting in May 2000, the Reactor Generic Issue Review Panel decided that the generic issue scope should be limited to (1) loads of approximately 27 metric tons (30 tons) or greater (designated as "very heavy"), and (2) commercial operating nuclear power plants. A representative sample of crane operating experience was obtained from nine nuclear power plant facilities consisting of 19 individual power plants which represents an approximate 13 percent sample of total U.S. nuclear power plant operating experience. A database was created, which estimated (based on this sample) the number of very heavy loads lifts by industry.

3.1 Pilot Plants for Crane Program and Operating Experience Reviews

Since many hardware and programmatic changes took place with the advent of NUREG-0612 in 1980, it was determined that this crane study should include only crane operational experience since that time. From January 1980 through December 2002, U.S. nuclear power plants have operated for a combined time more than 2000 years. The crane operating experience sample included plants of varying designs and ages. Most were multi-unit facilities, allowing more lift data to be retrieved. Table 7, "Pilot plants for crane program and operational experience reviews," lists the facilities visited.

Table 7: Pilot plants for crane program and operational experience reviews

Plant	Design Type	MWt	Commercial Operation Date	Onsite Visit Date
Brown's Ferry Units 1,2,3	BWR-Mark 1, GE 4, (AE) TVA	3293 3293 3293	1974 1975 1977	9/14-9/15/2000
Comanche Peak Units 1,2	PWR-Dry ambient, Westinghouse 4 Loop, (AE) Gibbs and Hill	3411 3411	1990 1993	11/27-11/29/2000
Diablo Canyon Units 1,2	PWR-Dry ambient, Westinghouse 4 Loop, (AE) PG&E	3411 3411	1985 1986	9/21-9/22/2000
Dresden Units 2,3	BWR-Mark 1, GE 3, (AE) S&L	2527 2527	1970 1971	7/11-7/13/2001
Grand Gulf	BWR-Mark 3, GE 6, (AE) Bechtel	3833	1985	12/11-12/13/2000
Limerick Units 1,2	BWR-Mark 2, GE4, (AE) Bechtel	3458 3458	1986 1990	12/4-12/5/2000
Oconee Units 1,2,3	PWR-Dry ambient, B&W, (AE) Bechtel	2568 2568 2568	1973 1974 1974	9/27-9/28/2000

Table 7: Pilot plants for crane program and operational experience reviews (Continued)

Oyster Creek	BWR-Mark 1, GE 2, (AE) Brown and Root	1930	1969	8/21-8/22/2000
Palo Verde Units 1,2,3	PWR-Dry ambient, CE80, (AE) Bechtel	3800 3876 3876	1986 1986 1988	11/15-11/17/2000

3.2 Very Heavy Load Crane Operating Experience at Pilot Plants

Table 8, "Total number of lifts with very heavy loads," lists crane lift data obtained from the eight pilot facilities dating back to the time that each plant received its operating license, or 1980, whichever was limiting up through the time of the onsite visit. The crane lifts shown do not include the crane lifts performed during the construction period of the plants. The data was retrieved from the pilot plants were obtained through actual searches of crane lift records, or by reviewing the typical number of lifts performed during routine outages and special outages. Items lifted include both safety and nonsafety related components. The total number of very heavy load lifts for the nine pilot facilities was approximately 7600.

Table 8: Total number of lifts with very heavy loads

Facility	Number of very heavy load lifts
Brown's Ferry 1,2,3	980
Comanche Peak 1,2	230
Diablo Canyon 1,2	344
Dresden 2,3	554
Grand Gulf	118
Limerick 1,2	950
Oconee 1,2,3	1656
Oyster Creek	504
Palo Verde 1,2,3	2277

3.3 Estimated Crane Operating Experience at US Nuclear Power Plants

To estimate the total number of lifts greater than approximately 27 metric tons (30 tons) for all U.S. nuclear power plants, it was necessary to normalize Table 8 lift data, taking into consideration how many refueling cycles had occurred, and the design type of the plant. The number of lifts per refueling cycle for each design type was then used to estimate the number of lifts occurring at the remaining power plants having a similar design. The total number of estimated very heavy load lifts for all US nuclear power plants that operated from 1980 through December 2002 was approximately 54000. 1980 was chosen because NRC guidance on crane operation was provided in 1980 through the publication of NUREG-0612.

3.4 Load Slips Involving Very Heavy Loads

Of the estimated 54000 very heavy load lifts at operating facilities following the issuance of NUREG-0612 in 1980, there were six very heavy load slips. A load slip is an uncontrolled vertical movement of a load that appears to be intermittent. None of the six very heavy load events resulted in radiation releases, risks to licensee personnel or the public. Table 9, “Load slips involving very heavy loads at operating nuclear plants (1980-2002),” provides a brief summary of each event. Figure 19, “Very heavy load slip distribution,” shows the distribution of very heavy load slips from 1969 through 2002, and includes one additional very heavy load slip which occurred at Dresden in 1976, which was prior to issuance of NUREG-0612. The Dresden event resulted in a load slip of 38 centimeters (15 inches), followed by a second slip while lowering the reactor pressure vessel head onto the reactor vessel using the reactor building crane which was reported to be single-failure-proof. The Comanche Peak Unit 1 very heavy load slip event was probably the most significant, involving the slip of a reactor coolant pump motor of 4.6 to 6.1meters (15 to 20 feet) in 1999, but was declared to be of very low risk significance through the Significance Determination Process. This event is discussed in greater detail in Section 2.3.8.

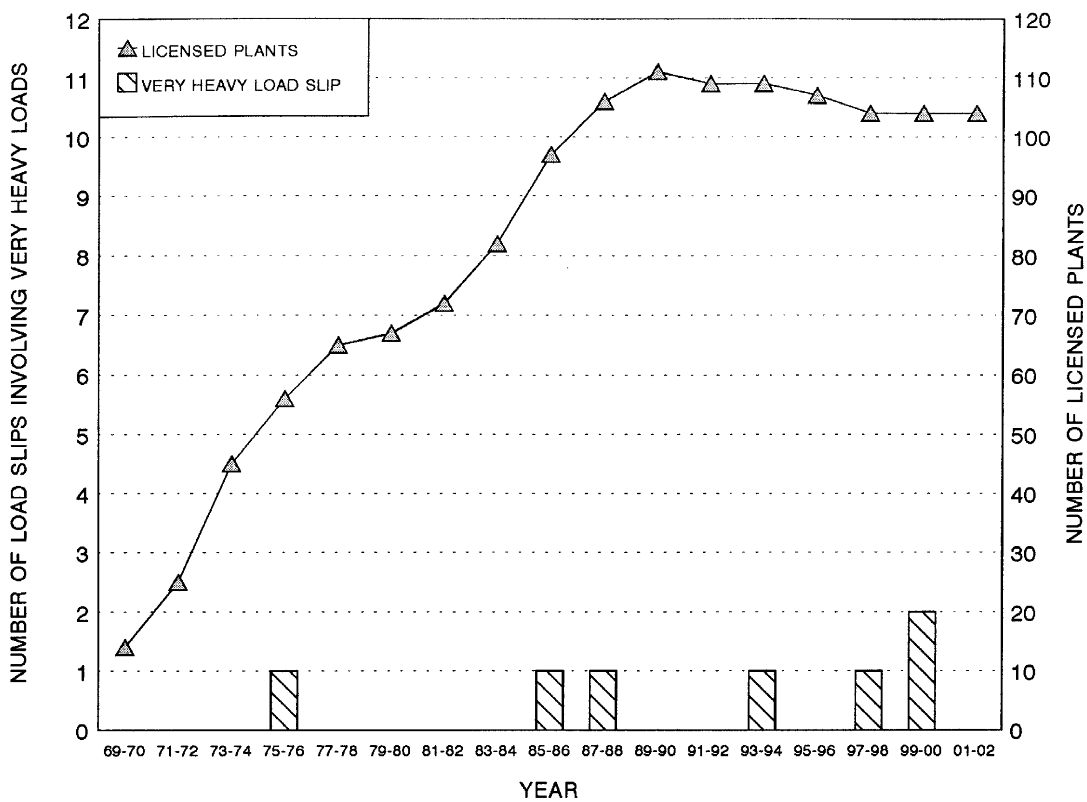

Figure 19: Very heavy load slip distribution

Table 9: Load slips involving very heavy loads at operating nuclear plants (1980-2002)

Plant	Event Date	Description
St. Lucie 1	Nov. 1985	While performing a lift of the upper guide structure weighing approximately 45 metric tons (50 tons), a bolt used to help secure a rigging device failed because it lacked proper thread engagement. The upper guide structure tilted approximately 15 centimeters (6 inches) when the bolt failed. No damage was done.
Fort Calhoun	April 1990	While lowering the reactor head, the load shifted, resulting in bending two alignment pins, and scratching the head flange.
Arkansas Nuclear One-1	Sept. 1993	While resuming a vertical lift of the reactor head, the head lowered instead of raising.
Byron	Dec. 1997	During the lift of a steam generator replacement runway section weighing approximately 26 metric tons (28.5 tons) located outside of containment, the runway section slipped about 4.6 meters (15 feet), came to an abrupt stop which caused the nylon rigging straps to fail. The runway section fell approximately 18 meters (60 feet) to the ground. Operator error was the most likely cause of the drop.
Comanche Peak 1	Oct. 1999	The Unit 1 reactor coolant pump motor weighing approximately 38 metric tons (42 tons) slipped approximately 4.6 to 6.1meters (15 to 20 feet) before coming to a stop during a lift using a nonsingle-failure-proof auxiliary hoist rigged to the polar crane. The slip occurred when the auxiliary hoist gearbox failed.
Crystal River	Nov. 1999	While using the polar crane, the lifting device for the reactor plenum was not attached properly and the load became cocked.

3.5 Load Drops Involving Very Heavy Loads

Of the estimated 54000 very heavy load lifts, there were three load drops. Figure 20, "Very heavy load drop distribution," shows the distribution of very heavy load drops from 1969 through 2002. A load drop is defined as an uncontrolled lowering of a load to the point where contact with the floor or some object stops any further decent. The figure shows seven very heavy load drops, four of which occurred during plant construction. The three very heavy load drop events that occurred after NUREG-0612 was issued and when the plant had an operating license, occurred because of human error, and ultimately because of rigging deficiencies and not because of crane deficiencies. The three events also did not occur near any safety related areas, and none resulted in radiation releases, risks to licensee personnel, or the public. Table 10, "Load drops involving very heavy loads at operating nuclear plants (1980-2002)," provides a brief summary of the three events. The Byron very heavy load drop event occurred while operating a mobile crane, while the San Onofre 3 and Turkey Point 4 very heavy load drop events occurred while operating turbine building overhead cranes.

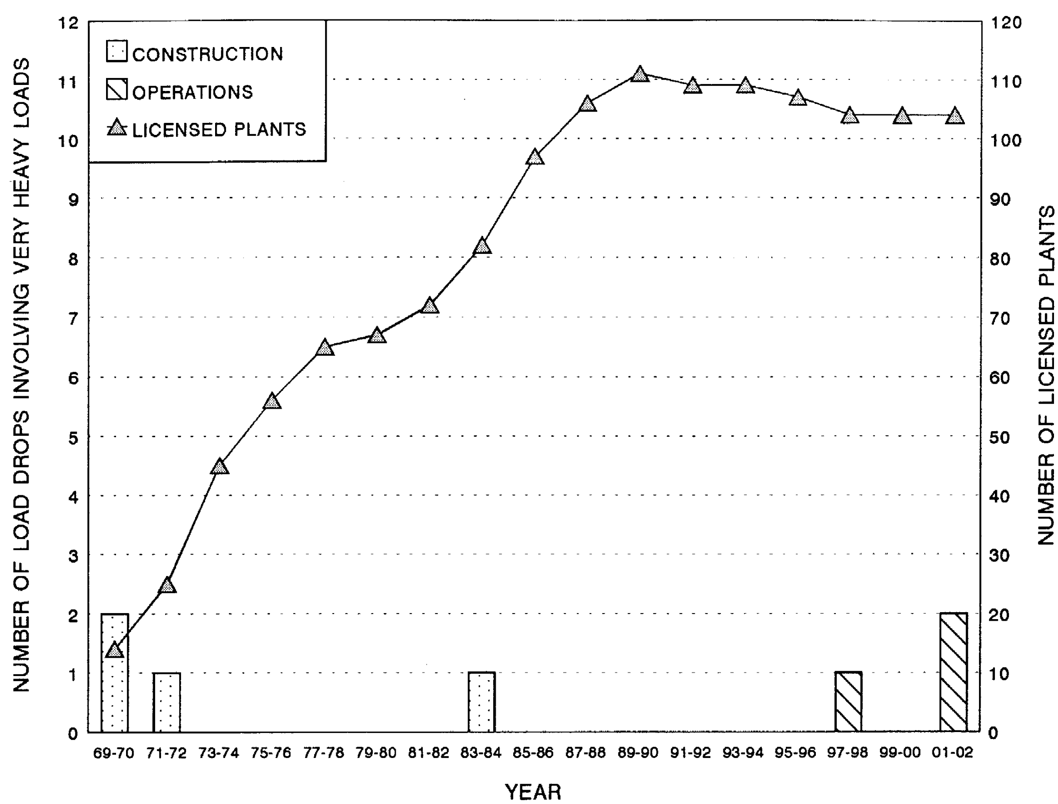

Figure 20: Very heavy load drop distribution

Table 10: Load drops involving very heavy loads at operating nuclear plants (1980-2002)

Plant	Event Date	Description
Byron	Dec. 1997	During the lift of a steam generator replacement runway section weighing approximately 26 metric tons (28.5 tons) located outside of containment, the runway section slipped about 4.6 meters (15 feet), came to an abrupt stop which caused the nylon rigging straps to fail. The runway section fell approximately 18 meters (60 feet) to the ground. Operator error was the most likely cause of the drop. This lift was by a mobile crane and the drop occurred outside of the power block.
San Onofre 3	May 2001	A mobile crane weighing approximately 34 metric tons (37.5 tons) was being lowered by the turbine building overhead crane when it was dropped approximately 12 meters (40 feet) to the Unit 3 turbine bay floor. The cause of the drop was failure of Kevlar slings because of the use of inadequate rigging softener material. The mobile crane was severely damaged.
Turkey Point 4	June 2001	A Link-Belt mobile crane weighing 34 metric tons (37.5 tons) was being raised by the turbine building overhead crane when is dropped approximately 20 centimeters (8 inches) to the Number 4 turbine building laydown area. The drop was caused by the failure of a Kevlar sling that was not properly protected at sharp corners. The mobile crane was inspected and found not to be damaged.

3.6 No Accident Sequence Precursor Events Involving Cranes

There were no Accident Sequence Precursor (ASP) classified events for the period 1985 through 2002 that involved a crane event. The ASP program identifies and categorizes precursors to potential severe core damage accident sequences. Accident sequences are those that, if additional failures occurred, could have resulted in inadequate core cooling, causing severe core damage. The ASP program analyzes potential precursors and calculates their conditional core damage probability (CCDP). The CCDP is the probability that the event or condition could have progressed to core damage given the existence of the failed or degraded protective or mitigating features or initiating event. To be classified as an ASP event, the event must have a CCDP of at least 1.0E-06. The most potentially risk-significant crane events involved a loss or partial loss of power. For the period 1985 through 2002,Table 5 summarizes the nine mobile crane events.

3.7 Load Drop Analysis and Potential Consequences

A generic load drop event tree was established to identify various accident sequences, and to quantify the sequence frequencies. The initiating event is the load drop. As shown in Section 4, none of the load drop calculations obtained from the participating nuclear power plant facilities discussed the consequences of heavy load drops on plant operations, or load drop collateral damage done to other plant components, such as safe shutdown equipment (SSE), needed for accident mitigation.

3.7.1 Load Drop Event Tree

Figure 21, “Load drop event tree,” provides a generic load drop event tree for those very heavy load drops that may impact SSE. An “endstate” of “OK” indicates load drop sequences that would not damage more than one train of SSE. An endstate of “Challenged” indicates sequences that would damage an entire system, which would require a backup or redundant system to be operable. Figure 21 also shows the estimated failure probabilities for each branch in the event tree. An explanation of the top events is provided below. A listing of the accident sequence failure rates, including a brief explanation is provided in Section 3.7.2.

<u>Number of very heavy load lifts per reactor year</u>

The approximate number of very heavy load lifts per reactor year (25) was calculated by taking the total number of very heavy load lifts (54000 lifts) that occurred since 1980 or commercial operation, which ever was the latest, and dividing it by the total number of reactor years for the same set of power plants having an operator license (over 2300 reactor years). This value was then used as the starting point for other branch event probabilities as discussed in this section.

<u>Load Drop</u>

For very heavy loads occurring at plants having an operating license, and after the issuance of NUREG-0612 in 1980, there were three load drops. The three very heavy load drops did not occur near any safety related areas, and none resulted in radiation releases, risks to licencee personnel, or the public. One event happened while operating a mobile crane, the other two occurred while operating turbine building cranes. All three were caused by rigging failures. Assuming that the number of very heavy load lifts was approximately 54000, the load drop frequency (drops/number of lifts) was calculated to be approximately 5.6E-05 (3/54000 lifts).

<u>Drop Over Safe Shutdown Equipment (On Level)</u>

The probability of a drop over SSE would be related to the probability of the failure to follow procedures. As shown in Figures 2 and 6, a large percentage of crane issues are either related to not following procedures, or to not properly implementing procedures. For the purposes of this assessment, it was conservatively assumed that all failures to follow procedures (159) event could have caused a drop over SSE. This would result in a probability of 159/54000 or approximately 3E-03 failures per lift. A lower estimate of 1.5E-03 was used which assumes that 50 percent of the crane events caused by not following procedures, resulted in a load drop over SSE.

<u>Safe Shutdown Equipment System Failure (On Level)</u>

For a very heavy load to be moved using a load path that went over SSE (on the same level that the heavy load was on) that was needed for safe operation or accident mitigation (e.g., both trains are damaged or an entire system is damaged) should be very low, however, this study uses a value between 5 and 10 percent. This value is a very conservative estimation of the loss of an SSE system because of separation of system trains and system diversity.

Floor Breach

Since those licensees that were visited as part of this study (see Table 7) had procedural load lift height guidance for differing load weights, and routine guidance to minimize the load lift height, a floor breach would seem very unlikely unless the crane operator failed to follow established procedures. The probabilities for each of the three branches in Figure 21 that would involve a floor breach are different. A factor of 5 separates each of the three floor breach pathways. The logic for the factor was the degree of crane operator error and plant operations error during the load lifts, caused by the potential common mode or common cause influences. Not only would the crane operator have to not follow procedures, but plant operations would have to disregard system alignment and operability requirements during the load lift. Consequently, the probability for a floor breach varied by a factor of 10 from the “best case” to the “worst case.” For this study, the first branch ranged between 1 and 5 percent, the second branch ranged between 2 and 10 percent, and the third branch ranged between 10 and 50 percent.

Safe Shutdown Equipment (Below Level)

Depending upon the load path, there may be SSE below the level over which the load would be transported. This could be in the form of controlling electrical, instrumentation, or mechanical fluid systems. The investigative level of this study (which was cursory) did not discover situations where redundancy or diversity would be eliminated. For the purposes of this study, the probability that SSE exists below level was conservatively assumed to range between 20 and 50 percent. The higher probability value (50 percent) shown in the worst case pathway was chosen because of potential common cause failures due to other preceding failures in the same pathway.

Safe Shutdown Equipment System Failure (Below Level)

Transporting very heavy loads over equipment that would be needed (e.g., both trains are damaged or an entire system is damaged) for plant accident mitigation would not be a conservative practice. This scenario is once again related to judgement or performance errors on the part of the crane operator or on plant operators. NUREG-0612 estimates that the probability of failure to follow a given procedure is between 1 and 5 percent. For this survey, 48 load path violations (for all load weights) that were reported in NRC or licensee documents, which is much less than one percent of the number of very heavy load lifts that were performed. Of the estimated 54000 very heavy load lifts, none resulted in a SSE system failure. For the purposes of this study, the probability is conservatively assumed to be double that of NUREG-0612, resulting in probabilities between 2 and 10 percent.

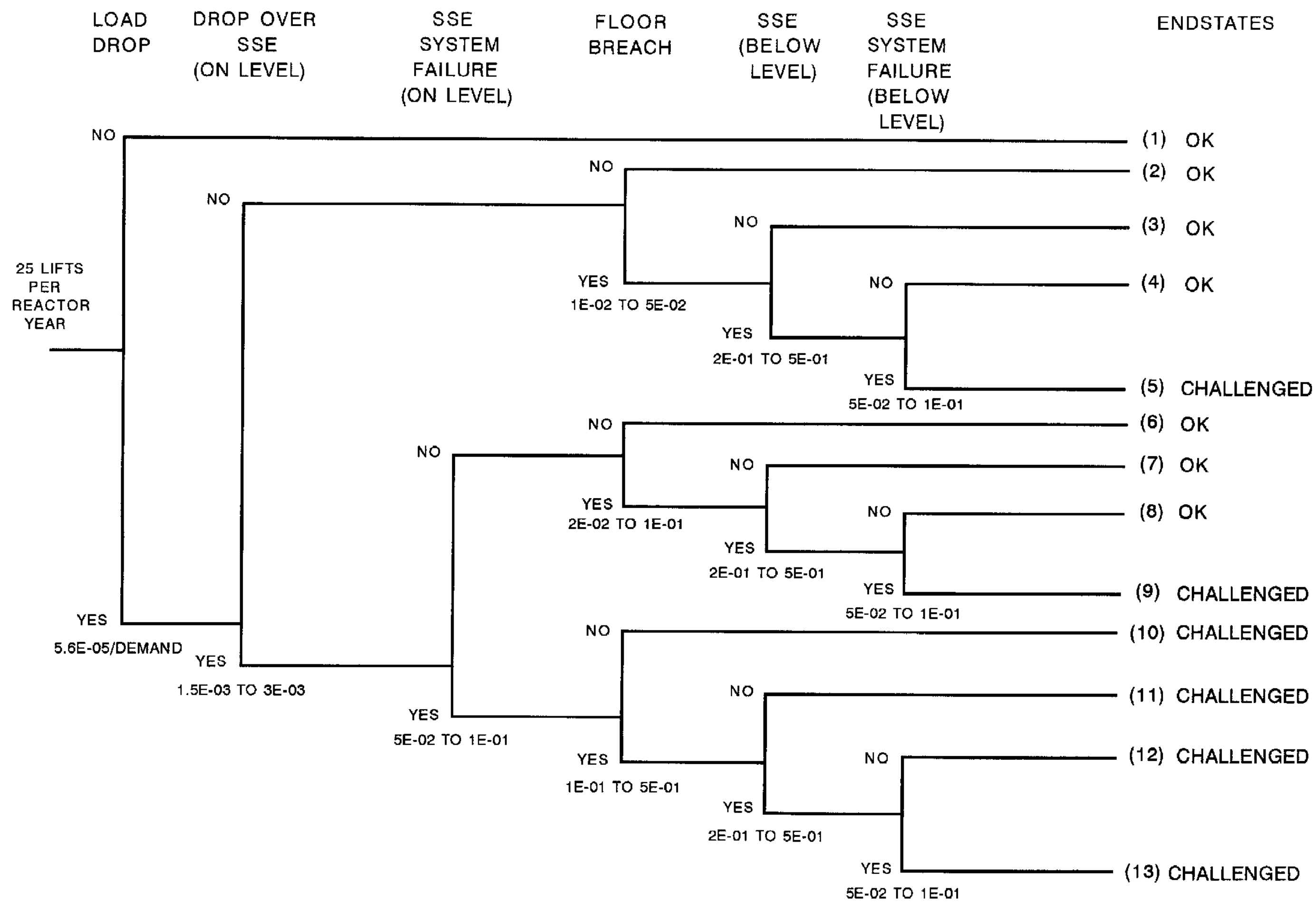

Figure 21: Load drop event tree

3.7.2 Potential Consequence of a Very Heavy Load Drop

Several sequences were established, and probabilities of each branch were estimated based on operational experience or information contained in NUREG-0612 which in turn references WASH-1400, "Reactor Safety Study." Since plant visits contained in this survey did not include a design review to determine electro-mechanical separation vulnerabilities for redundant or diverse systems of SSE, conservative probabilities were used to estimate the failure probabilities. For many older nuclear plants, safety-related electrical and mechanical systems were "field run" meaning that isometric design or as-built drawings do not exist showing the layout. Consequently, detailed site-specific information was not used to evaluate what equipment may be lost in the event of a load drop at various plant locations. Walkdowns of critical areas were accomplished to determine any obvious system separation concerns. Crane operating experience to determine failure frequencies, in conjunction with human reliability data, was used to estimate the equipment failure frequencies or failure probabilities of operator actions. Table 11, "Potential consequences of very heavy load drops," lists the endstate (see figure 21), accident frequency per reactor year, and the plant consequence.

<u>BWR v.s. PWR</u>

Very heavy load drops in BWR plants are more risk significant than very heavy load drops in PWRs, in that, for PWRs, spent fuel cask transfer occurs in an area separate from the reactor building and many safety related systems. However, for BWRs, many very heavy loads would be lifted and moved on the upper floor of the reactor building. Should a floor breach occur during a load drop, there are many safety-related components located on lower floors which could be disabled. Because of the vast differences between reactor safety system layout even within the same design type (i.e., BWR v.s. PWR, or NSSS vendor) more exact consequence analysis of very heavy load drops at different locations within a nuclear plant was not practical for this report. Even given the many NRC generic communications on heavy load concerns, few licensees have performed a consequence analysis of heavy load drops as shown in Table 12. Of the 74 facilities listed on Table 12, eight licensee responses to Bulletin 96-02 indicated that a consequence analysis had been done at their facility for heavy load drops.

Table 11: Potential consequences of very heavy load drops

Endstate	Accident Frequency per Reactor Year[1]	Plant Consequence	Plant Status
1	No load drop path	None. No load drop occurs.	OK
2	1.4E-03 (mean)	Load drop occurs, but does not result in any train or system damage.	OK
3	2.8E-06 to 3.5E-05	Load drop occurs, resulting in a floor breach, but does not result in a SSE train or system damage.	OK
4	2.8E-05 to 3.5E-05	Load drop occurs, resulting in a floor breach, and one SSE train disabled	OK
5	1.4E-07 to 3.5E-06	Load drops occurs, resulting in a floor breach, and one SSE system disabled.	Plant is challenged.[2]
6	2.1E-06 to 4.2E-06	Load drop occurs, resulting in one SSE train being disabled. No floor breach or other damage to SSE.	**OK**
7	8.4E-09 to 2.1E-07	Load drop occurs, resulting in one SSE train being disabled. A floor breach occurs, but no other SSE damage occurs.	OK
8	8.4E-09 to 2.1E-07	Load drop occurs, resulting in one train disabled, a floor breach and one additional SSE train disabled in another system (both systems remain intact).	OK
9	4.2E-10 to 2.1E-08	Load drop occurs, resulting in one SSE train disabled, a floor breach, and one SSE system disabled.	Plant is challenged.
10	1.1E-07 to 4.2E-07	Load drop occurs, resulting in one SSE system disabled with no floor breach.	Plant is challenged.
11	2.1E-09 to 1.1E-07	Load drop occurs, resulting in one SSE system disabled, a floor breach, but no other train or system damage.	Plant is challenged.
12	2.1E-09 to 1.1E-07	Load drop occurs, resulting in one SSE system disabled, a floor breach, and one other SSE train damaged.	Plant is challenged.
13	1.1E-10 to 1.1E-08	Load drop occurs, resulting in two systems disabled, including a floor breach.	Plant is challenged.

[1]Assumes an average of 25 very heavy loads per reactor year.
[2]A condition where at least one SSE system has been disabled because of a load drop.

4 LICENSEE VERY HEAVY LOAD DROP CALCULATIONS

Load drop calculations involving very heavy loads were obtained from each facility shown in Table 7. For selected calculations, their basic scenario, assumptions, and predicted consequence is provided in Appendix F, "Heavy Load Drop Calculations at U.S. Nuclear Power Plants." A review of the load drop calculations indicated that calculational methodologies, basic accident scenarios, assumptions, and predicted consequence varied greatly from licensee to licensee, producing different consequences. Accurate load drop analysis is essential, since each licensee uses load drop calculations to determine transport height restrictions which are referenced in their heavy load lift procedures. Load drop analyses also determine locations where other measures besides load height restrictions are necessary (e.g., impact limiting devices, interlocks to prevent crane motion over certain areas, or employment of single-failure proof handling systems).

4.1 Load Drop Calculation Assumptions

The load drop assumptions varied greatly from facility to facility. The assumptions made in most calculations included (1) the postulated load, (2) weight of load, (3) drop and impact configuration, (4) drop height, (5) target composition including material, thickness, and reinforcement, (6) striking velocity, (7) impact geometry, and (8) ductility ratios. Rather than assuming a drop height, some calculations worked backwards from an unacceptable result (i.e., failure of the target) to determine the maximum allowable drop height or weight.

4.2 Load Drop Consequences

Given the wide variety of load drop scenarios presented in Appendix F, an equally wide variety of consequences was also expected. However, similar load drop scenarios also produced widely differing consequences. For example, a sample of load drop weight, height, and consequences are provided for plants that were visited as part of this survey of crane operating experience.

<u>Oyster Creek</u>

A calculation predicted that the maximum allowable drop height for a fuel cask weighing approximately 41 metric tons (45 tons) over a reinforced concrete slab that was 41 centimeters (16 inches) thick, was 7 centimeters (2.77 inches).

<u>Brown's Ferry</u>

A calculation predicted that a load weighing 91 metric tons (100 tons) could drop on a hypothetical reinforced concrete slab that was 46 centimeters (18 inches) thick from a height of 0.9 meter (3 feet) and not penetrate the slab.

<u>Limerick</u>

A calculation postulating the drop of a steam dryer assembly weighing approximately 41 metric tons (45 tons) from a height of 1.8 meters (6 feet) for either a flat/distributed area contact or an edge contact) showed that stress levels were well within the allowable range.

Comanche Peak

A calculation showed that the maximum allowable drop height for a reactor coolant pump motor assembly weighing 38 metric tons (42.4 tons) over a reinforced concrete slab 66 centimeters (26 inches) thick, was 64 centimeters (25 inches).

Dresden

The maximum weight of a load dropped down the reactor building equipment hatch (a drop of 29 meters [95.5 feet]) to prevent scabbing of the 61 centimeters (24 inches) thick reinforced concrete slab was 0.9 metric ton (1 ton). To prevent perforation of the 61 centimeters (24 inches `) thick slab, the weight of the load could not exceed 5.2 metric tons (5.75 tons).

4.3 Load Path Control Variations

The facilities visited used varying means to create and maintain "restricted areas" and safe load paths for load movements. NUREG-0612 provides guidance concerning the establishment of safe load paths. Some licensees marked restricted areas using paint, some used detailed load handling procedures, while others used various interlocks to control crane movements. Heavy load drop analyses also help to determine locations where other measures, besides load height restrictions, are necessary (e.g., impact limiting devices, interlocks to prevent crane motion over certain areas, or employment of single-failure proof handling systems). Because of weaknesses in load drop calculations, and load height control, weaknesses were also discovered in load path control processes.

5 NRC GENERIC COMMUNICATIONS RELATED TO CRANE OPERATION

A review of generic communications (GCs) was completed as part of this survey of operating experience. Appendix G, "NRC Generic Communications Involving Crane Operating Experience," includes a chronological listing of generic communications involving crane operating experience, including a brief summary of the various issues. The 29 GCs that the NRC issued between 1976 and 2002 were sorted into seven basic categories. In most areas, crane operating performance has improved or remained constant.

5.1 Numerous Generic Communications Involving Crane Operation

There have been 29 NRC GCs that have involved load movements at U.S. nuclear power plants dating back to 1976. There have been seven basic categories of GCs related to crane operation. Category 1, with nine GCs, is considered the most important and discusses heavy loads moved on the refueling floor, load drop analysis for heavy loads, identification of heavy loads that are lifted over safe shutdown equipment, and the consequence of a load drop on selected equipment. Most of the GCs were issued to inform licensees of important operating experience information. A few GCs (issued as generic letters) requested licensees to provide information on their crane programs for NRC evaluation. Ranking the seven categories by the number of generic communications generated (including supplements) in each category produces the following:

(1) Load drops, load paths, and handling controls (Nine GCs)

(2) Fuel movements and programmatic controls (five GCs)

(3) Fuel assembly damage (five GCs)

(4) Dropped fuel assemblies (four GCs)

(5) Miscellaneous overhead crane problems (three GCs)

(6) Lifting device issues (two GCs)

(7) Mobile crane issues (one GC)

5.2 Generic Communications Requesting Licensee Action

The generic communication process has been used to request similar information from licensees regarding heavy load movements in PWRs and BWRs from 1978 through 1996. A few GCs (issued as generic letters and one bulletin) requested licensees to provide information on their crane programs for NRC evaluation. The accuracy and consistency of the information received is questionable. Accumulated licensee information relating to crane type, load drop analysis results, or consequence analysis was still incomplete following licensee responses to Bulletin 96-02 which was closed out in 1998. The accuracy and consistency of the information received by the NRC is questionable. Summarized information that was requested from licensees is included in Sections 5.2.1 through 5.2.5.

5.2.1 Generic Letters 78-15, 78-16, and 78-17

Generic Letters 78-15, 78-16, and 78-17 alerted licensees to review their current procedures for the movement of heavy loads over spent fuel to assure that the potential for a handling accident which could result in damage to spent fuel is kept at a minimum. Each of these three generic letters has the same licensee action requested, as follows:

(1) Provide a diagram which illustrates the physical relation between the reactor core, the fuel transfer canal, the spent fuel storage pool and the set down, receiving or storage areas for any heavy loads moved on the refueling floor.

(2) Provide a list of all objects that are required to be moved over the reactor core (during refueling), or the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram of the movement path utilized (including carrying height) and the frequency of movement.

(3) What are the dimensions and weights of the spent fuel casks that are or will be used at your facility?

(4) Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC staff.

(5) Identify any heavy loads that are carried over equipment required for the safe shutdown of a plant that is operating at the time the load is moved. Identify what equipment could be affected in the event of a heavy load handling accident (piping, cabling, pumps, etc.) and discuss the feasibility of such an accident affecting this equipment. Describe the basis for your conclusions.

(6) If heavy loads are required to be carried over the spent fuel storage pool or fuel transfer canal at your facility, discuss the feasibility of a handling accident which could result in water leakage severe enough to uncover the spent fuel. Describe the basis for your conclusions.

(7) Describe any design features of your facility which affect the potential for a heavy load handling accident involving spent fuel, e.g., utilization of a single failure-proof crane.

(8) Provide copies of all procedures currently in effect at your facility for the movement of heavy loads over reactor core during refueling, the spent fuel storage pool or equipment required for the safe shut-down of a plant that is operating at the time the move occurs.

(9) Discuss the degree to which your facility complies with the eight (8) regulatory positions delineated in Regulatory Guide 1.13 (Rev. 1, December 1975) regarding Spent Fuel Storage Facility Design Basis.

5.2.2 Generic Letter 80-113

Generic Letter 80-113 (originally unnumbered) requested licensees to review their controls of handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 were satisfied, and to make any needed changes. Licensees were requested to provide the following:

(1) Submit a report documenting required changes and modifications, and how the guidelines of NUREG-0612 will be satisfied.

(2) Furnish confirmation within six months that implementation of those changes and modifications you find are necessary will commence as soon as possible without waiting for staff review, so that all such changes, beyond the above interim actions, will be completed within two years of submittal.

(3) Furnish justification within six months for any changes or modifications that would be required to fully satisfy the NUREG-0612 guidelines you believe are not necessary.

5.2.3 Generic Letter 81-07

Generic Letter 81-07 requested licensee to review their controls of handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 were satisfied, and to make any needed changes. This request supplemented those in Generic Letter 80-113. Licensees were requested to provide the following:

(1) 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.

(2) 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.

(3) 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 consideration of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects.

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

5.2.4 Generic Letter 85-11

Generic Letter 85-11 rescinded the requirement to perform a Phase II review requested in Generic Letter 80-113 and Generic Letter 81-07. Phase II addressed the need for electrical interlocks/mechanical stops, or alternatively, single-failure proof cranes or load drop analyses in the spent fuel pool area (PWR), containment building (PWR), reactor building (BWR), other areas and the specific guidelines for single-failure-proof handling systems. The generic letter stated that based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action is not required to reduce the risks associated with the handling of heavy loads. The generic letter also stated that a detailed Phase II review of heavy loads is not necessary and Phase II is considered completed. A cost benefit analysis for upgrading polar cranes to single failure proof (included as Attachment I to the generic letter) indicated that the NRC could not perceive a significant enough benefit in the conversion to a single failure proof polar crane to warrant the high costs estimated.

5.2.5 Licensee Response to NRC Bulletin 96-02

NRC Bulletin 96-02 was initiated because of the planned movement of 100 ton dry storage casks by Oyster Creek. Based on the NRC audit of Oyster Creek's 10 CFR 50.59 evaluation of cask movement, the staff was concerned that other licensees may believe that their heavy load operations were in compliance with the regulations, because they had completed Phase I of the generic letter of December 22, 1980, and the closeout or discontinuation of Phase II through Generic Letter 85-11. In addition, Generic Letter 85-11 concluded that the risks associated with damage to safety-related equipment were relatively small because (1) nearly all load paths avoid this (safety-related) equipment, (2) most equipment is protected by an intervening floor,

(3) there is redundancy or diversity of components, and (4) crane failure probability is generally independent of safety-related systems.

NRC Bulletin 96-02 requested licensees to provide the staff with specific information relating to their heavy loads program and plans within 30 days. Many of the licensees that responded to the bulletin, provided incomplete information relating to crane types, load drop analysis, consequence analysis, plant status during movement, and crane type to be used for the load movements. Eight respondents indicated that a consequence analysis had been done at their facility for heavy load drops. Table 12, "Licensee response to NRC Bulletin 96-02," provides a compilation of licensee responses.

Table 12: Licensee response to NRC Bulletin 96-02

Plant	Crane Type	Plant Status at Load Movement	Load Drop Analysis	Consequence Analysis
Arkansas Nuclear One	Meets NUREG-0612	At power	Yes	Yes
Beaver Valley 1,2	Not specified	At power, some loads over safety-related equipment	Not specified	Not specified
Big Rock Point	Not specified	Shutdown	Not specified	Not specified
Brown's Ferry 1,2,3	Not specified	At power	No	No
Brunswick 1,2	Meets NUREG-0612	Shutdown	No	No
Braidwood 1,2	Not specified	Not specified	No	No
Byron 1,2	Not specified	Not specified	No	No
Callaway	Not specified	Shutdown	Yes	Not specified
Calvert Cliffs 1,2	Single-failure-proof	At power	Yes	Not specified
Catawba 1,2	Not specified	Shutdown	Yes	Yes
Clinton	Single-failure-proof	Shutdown	Yes	No specified
Comanche Peak 1,2	Not specified	Shutdown	Not specified	Not specified
Cook 1,2	Not specified	Not specified	No	No
Cooper	Not single-failure-proof	At power	No	No
Crystal River	Meets 0612 crane upgrade requirements	At power Shutdown for unreviewed loads	No	No
Davis-Besse	Single-failure-proof	At power	No	No
Diablo Canyon 1,2	Not specified	Not specified	Not specified	Not specified
Dresden 2,3	Single-failure-proof	Shutdown	No	No
Duane Arnold	Not single-failure-proof	At power	No	Yes
Farley 1,2	Not specified	Shutdown	No	No
Fermi	Single-failure-proof	At power	No	Not specified
Fitzpatrick	Not specified	At power (not at power for casks)	No	No

Table 12: Licensee response to NRC Bulletin 96-02 (Continued)

Plant	Crane Type	Plant Status at Load Movement	Load Drop Analysis	Consequence Analysis
Fort Calhoun	Not single-failure-proof	Shutdown	Yes	Not specified
Ginna	Not single-failure-proof	Not specified	Not specified	Not specified
Grand Gulf	Not single-failure-proof	At power	No	Not specified
Haddam Neck	Not specified	Shutdown	Not specified	Not specified
Harris	Not specified	Not at power for unreviewed loads Shutdown for other loads	Yes	Not specified
Hatch 1,2	Single-failure-proof	Cask dry runs at power	No	No
Hope Creek	Single-failure-proof	At power	No	No
Indian Point 2	Not specified	Not specified	Not specified	Not specified
Indian Point 3	Not specified	Not specified	Not specified	Not specified
Kewaunee	Not specified	Some at power	Yes	Possibly
LaSalle 1,2	Single-failure-proof	Shutdown	No	No
Limerick 1,2	Single-failure-proof	Low power	No	No
Maine Yankee	Not specified	Not specified	Not specified	Not specified
McGuire 1,2	Not specified	Some at power	No	No
Millstone 1	Not specified	Shutdown	Some in FSAR	Not specified
Millstone 2	Not specified	Shutdown	Yes	Not specified
Millstone 3	Not specified	Shutdown for unreviewed loads Others loads at power	Not specified	Not specified
Monticello	Single-failure-proof	At power	Reference basis	No
Nine Mile Point 1	Single-failure-proof	At power	No	No
Nine Mile Point 2	Single-failure-proof	At power	No	No
North Anna 1,2	Not specified	Shutdown	Yes	Yes
Oconee 1,2,3	Not specified	Shutdown	Yes	Yes
Oyster Creek	Not single-failure-proof	At power	No	No, not credible
Palisades	Not specified	Not specified	No	No
Palo Verde 1,2,3	Meets NUREG-0612 upgrade requirements	Shutdown	No	No
Peach Bottom 2,3	Single-failure-proof	Low power	No	No
Perry	Not specified	Shutdown	No	No
Pilgrim	Not specified	Not specified	No	No
Point Beach 1,2	Single-failure-proof	Not specified	No	No
Prairie Island 1,2	Meets NUREG-0612 upgrade requirements	Shutdown	No	No
Quad Cities 1,2	Single-failure-proof	Shutdown	No	No

Table 12: Licensee response to NRC Bulletin 96-02 (Continued)

Plant	Crane Type	Plant Status at Load Movement	Load Drop Analysis	Consequence Analysis
River Bend	Meets NUREG-0612 upgrade requirements	Shutdown	No	No
Robinson	Meets NUREG-0612 upgrade requirements	Shutdown	Uses a lifting yoke which precludes the possibility of a drop accident	No
Salem 1,2	Not specified	Some at power	Not specified	No
San Onofre 2,3	Single-failure-proof	Some at power	Yes	No
Seabrook	Not specified	At power	Yes	Not specified
Sequoyah 1,2	Not specified	Not specified	In licensing basis	Not specified
South Texas 1,2	Meets NUREG-0612 upgrade requirements	Shutdown - fuel At power for other loads	Yes	Yes
St. Lucie 1,2	Not specified	At power	Not specified	Not specified
Summer	Not specified	Not specified	Yes	Yes
Surry 1,2	Not specified	Shutdown	Yes	Yes
Susquehanna 1	Single-failure-proof	At power	No	Not specified
Susquehanna 2	Not single-failure-proof	At power	No	Not specified
Three Mile Island	Not specified	Some at power, not over fuel or more than one train of safety-related equipment	Yes	Not specified
Turkey Point 3,4	Meets NUREG-0612 upgrade requirements	Not specified	Yes	Not specified
Vermont Yankee	Single-failure-proof	At power	Yes	Not specified
Vogtle 1,2	Not specified	Only move previously analyzed loads	Not specified	Not specified
Washington Nuclear 2 (Columbia)	Meets NUREG-0612 upgrade requirements	Not specified	Not specified	Not specified
Waterford	Not specified	Some at power, interlocks prevent movement over fuel	Yes	No
Watts Bar	Not specified	Some at power	No	No
Wolf Creek	Not specified	Not specified	Yes	Not specified
Zion 1,2	Not specified	Not specified	No	No

6 HEAVY LOAD MOVEMENTS AND CRANE CLASSIFICATION

6.1 Single Failure Proof Crane Guidance

NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” and NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” provide current NRC guidance for what constitutes design requirements for single-failure-proof cranes (NUREG-0554), or what modifications are required to upgrade an existing crane to a single-failure-proof classification (Appendix C of NUREG-0612). Both of these documents have been interpreted differently by licensees and vendors. It was also unclear what “credit” could be given by the NRC to licensees that had modified cranes to make them more reliable and failure proof, when making very heavy load movements over safety-related equipment, if the crane did not meet all of the design criteria of NUREG-0554 or Appendix C of NUREG-0612.

A third document relating to cranes used in nuclear service is ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” received ANSI approval in October 1998. The NOG-1 Standard applies to the design, manufacture, testing, inspection, shipment, storage, and erection of cranes (Types I, II, and III) covered by the Standard. NOG-1, Type I crane design criteria appears to be similar to design criteria in NUREG-0554. The definition of a Type I crane in the NOG-1 Standard is:

> a crane that is used to handle a critical load. It shall be designed and constructed so that it will remain in place and support the critical load during and after a seismic event, but does not have to be operational after this event. Single failure-proof features shall be included so that any credible failure of a single component will not result in the loss of capability to stop and hold the critical load.

NOG-1 defines a critical load as,

> any lifted load whose uncontrolled movement or release could adversely affect any safety-related system when such a system is required for unit safety or could result in potential off-site exposure in excess of the limit determined by the purchaser.

6.2 Crane Classification Issues

Although single-failure-proof cranes share many common design features (e.g., dual reeving, redundant limit switches, and redundant brakes), the remaining criteria for declaring a crane as single-failure-proof have been inconsistently applied. Crane manufacturers also stressed that NUREG-0554 was ambiguous in some areas, and that clarifications or changes needed to be made to both NUREG-0612 and NUREG-0554. Industry suggested that a preferred approach would be to consider adopting NOG-1, Type I (with minor changes) as an acceptable approach to meeting NUREG-0554 and for upgrading cranes to single-failure-proof status. NOG-1 contains much more specific design information than NUREG-0554 in explaining design criteria for single-failure-proof cranes.

In addition, while some licensees listed a crane as single-failure-proof, or that it met NUREG-0612 upgrade requirements, all the single-failure-proof design criteria listed in NUREG-0554 may not be fully met.

6.3 Preventable Load Drop Events With Single-Failure-Proof Cranes

As previously discussed in Section 2.3.2, most load drop events occurring during the period 1968 through 2002 have been the result of poor program implementation or human performance errors involving fuel assembly drops, mobile crane operation, or below-the-hook events, rather than the result of crane or hoist hardware issues. Many single-failure-proof crane design features compensate for human performance errors that have resulted in load drops, as well as hardware failures. Single-failure-proof handling system programmatic requirements for lifting devices reduce the potential for a single rigging error to cause a load drop. A review of very heavy load lift events occurring after the advent of NUREG-0612 shows that there were no very heavy load drop events that could have been prevented had only a single-failure-proof crane been employed in the lift. However, there were load or hook and block assembly drops that could have been prevented with the use of single-failure-proof cranes and lifting devices. For heavy load drops from1 to 27 metric tons (1.1 to 30 tons), there were two events that could have been prevented had a single-failure-proof crane been used. However, one event occurred at the Shoreham nuclear plant in 1993 while undergoing decommissioning, and involved a load weighing 4.5 metric tons (5 tons); the second event occurred at the Peach Bottom facility in 2002, and involved a load weighing 22 metric tons (24 tons). Neither of these two events posed a risk.

<u>Very Heavy Load Lifts (Greater than 30 Tons)</u>

The four very heavy load lifts that resulted in a drop (i.e., River Bend (BWR) in 1983, Byron (PWR) in 1997, San Onofre (PWR) in 2001, and Turkey Point (PWR) in 2001) all occurred outside of containment, the reactor building, or the spent fuel pool area, and were not in safety related areas.

- The River Bend event in 1983 involved the drop of a shield building dome form weighing approximately 363 metric tons (400 tons) that was being lifted by a mobile crane (the mobile crane mast buckled dropping the very heavy load 9 meters (30 feet). The unit was under construction at the time of the event.

- The Byron event in 1997 involved the drop of a runway section weighing slightly less than 27 metric tons (30 tons) for a steam generator replacement activity that was being lifted by a mobile crane outside of the containment.

- The San Onofre and Turkey Point events in 2001 load drops were both performed by turbine building cranes in a laydown area and were both caused by rigging failures, and not by crane hardware problems.

<u>Heavy Load Lifts Greater than 1 metric ton (1.1 tons) but less than 27 metric tons (30 Tons)</u>

Of the six heavy load lifts that resulted in a drop (i.e., Callaway in 1981, Shoreham in 1993, Fermi in 1994, Indian Point 2 in 1996, Susquehanna in 1997, and Peach Bottom 2 in 2002), four events involved either rigging failures or mobile crane operation. The remaining two load drop events (Shoreham and Peach Bottom 2) could have been prevented had single-failure-proof cranes been used, however, neither event posed any risk. The Shoreham event occurred during decommissioning, and the Peach Bottom 2 event was of very low risk significance.

- An event at the Callaway nuclear plant in 1981 involved the drop of a concrete hatch weighing 11 to 16 metric tons (12 to 18 tons) when a mobile crane boom collapsed on the service water building.

- An event at the Shoreham nuclear plant in 1993 involved the drop of a refueling jib crane that was suspended from a polar crane auxiliary hook that was not single-failure-proof. The event resulted in an industrial injury.

- An event at the Fermi nuclear plant in 1994 involved the drop of a resin liner weighing 8.2 metric tons (9.5 tons) when a mobile crane tipped over.

- An event at the Indian Point 2 in 1996 involving the drop of a transportation container weighing 2.3 metric tons (2.5 tons) when the rigging slings came off the crane hook.

- An event at the Susquehanna nuclear plant in 1997 involving the drop of a tool box weighing 1.8 metric tons (2 tons) while being lifted by the reactor building auxiliary hoist.

- An event at the Peach Bottom 2 nuclear plant in 2002 occurred when a recirculation pump motor weighing approximately 22 metric tons (24 tons), dropped to its stand when a hoist chain broke (the hoist was not single-failure-proof).

<u>Load Lifts Less than 1 metric ton (1.1 tons)</u>

There was no risk significance for load drops of components less than 1 metric ton (1.1 tons). These events were mostly fuel assembly drops, with a few below-the-hook or mobile crane events included. None of these events were risk significant, but did involve industrial accidents (e.g., three injuries).

7 CRANE OPERATING EXPERIENCE STUDIES

Several crane studies have been performed to estimate failure probabilities, component reliability, root causes, and human factors issues. NUREG-0612, along with more recent studies, are briefly discussed in Sections 7.1 though 7.6.

7.1 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"

NUREG-0612 was published by the Office of Nuclear Reactor Regulation (NRR) of the NRC in July 1980. This study was based on data available from (1) Occupational Safety and Health Administration (OSHA), involving root cause data on over 1000 crane accidents during an unspecified time period, (2) the Department of the Navy, involving 466 crane events occurring between February 1974 and October 1977, and (3) NRC Licensee Event Report involving 34 crane events occurring between July 1969 and July 1979. Multiple probabilities are given for various scenarios, however, the study states, "Based on the data collected from the Navy, it is expected that the probability of handling system failure for nuclear plant cranes will be on the order of between 1E-05 and 1.5E-04 per lift." This probability of failure was a best estimate since Navy crane data does not indicate how many lifts were actually performed, i.e., only the number of problems have been quantified.

7.2 EEG-74, "Probability of Failure of the TRUDOCK crane system at the Waste Isolation Pilot Plant (WIPP)"

Appendix B, "EEG-74: Probability of Failure of the TRUDOCK Crane System at the Waste Isolation Pilot Plant (WIPP)," includes a report by the Environmental Evaluation Group (EEG) of New Mexico that discusses the probability of the failure of the TRUDOCK crane system at the WIPP located in New Mexico. The study used the failure data in NUREG-0612 and other sources to estimate the mean failure rate. Based on their calculations, a mean failure rate of 9.7 E-03 (1/year) was obtained. The mean failure rate is based on combined equipment failure rate per demand of 5E-06, and a combined operator error rate of 1.7E-07 per demand. Calculations of confidence levels showed that there was a 71 percent likelihood that not more than one dropped load would occur in 103 years, and that there was a 95 percent likelihood that not more than one dropped load would occur in approximately 34 years. The EEG report predicted a much lower human error rate (e.g., a 25 percent contribution) than is experienced in U.S. Navy reports or in the commercial U.S. nuclear power plant industry. This lower human error rate for the WIPP is attributed to greater training. In contrast, for Navy crane operation, human error rates between 90 and 95 percent were reported for 1996, 1997, and 1998.

7.3 Department of Energy Study, "Independent Oversight Special Study of Hoisting and Rigging Incidents Within the Department of Energy"

Appendix C, "Independent Oversight Special Study of Hoisting and Rigging Incidents Within the Department of Energy [DOE]," includes the DOE special study which was issued in 1996. The DOE report presents the results of an analysis of hoisting and rigging (H&R) incidents, covering the period from 1993 to 1996. DOE defined H&R to include the raising, moving, and unloading of materials, either by large power-lifting equipment, such as cranes and forklifts, or by smaller, light duty manual and power-operated equipment, such as hoists, chainfalls, and block and tackle. Human error, whether directly associated with supervisors or equipment operators represented approximately 94 percent of H&R incidents. Factors not related to human performance, such as equipment failure and weather, were responsible for only 6 percent of H&R incidents. Inattention to detail (56 percent) and not following procedures (28 percent) accounted for 84 percent of H&R incidents caused by personnel error. The report analyzed 66 "relevant" hoisting and rigging incidents occurred during the 30 month study period. "Relevant" was defined as: (1) an event occurring during hoisting and rigging operations, or the use of hoisting and rigging equipment, as defined in the U.S. Department of Energy Hoisting and Rigging Handbook, AND (2) one that resulted in unsafe or improper conditions that necessitated the immediate suspension of the hoisting and rigging operation for any period of time, led to a near miss, or caused an accident. Unfortunately, no listing of the relevant crane incidents were given, however, root causes of the crane incidents were listed, and are shown in Table 13, "Root causes of crane incidents at DOE facilities." As seen by the table, most crane incidents at DOE facilities are related to human factors issues such as inattention to detail, work organization and planning, and programmatic areas rather than crane hardware failures or deficiencies.

Table 13: Root causes of crane incidents at DOE facilities

Root Cause	Percent	Root Cause	Percent
Inattention to detail	20	Other human error	3
Work organization and Planning	18	Insufficient refresher training	3
Procedure not used or used incorrectly	9	Lack of procedure	2
Policy not adequately defined, disseminated, or enforced	9	Communication problem	2
Defective or inadequate procedure	9	Inadequate work environment	0
Inadequate administrative control	9	Inadequate supervision	0
Inadequate or defective design	5	Error in equipment or materials selection	0
Defective or failed part	5	Weather	0
Insufficient practice or hands-on experience	5	No training provided	0
Other management problem	3		

7.4 California Department of Industrial Relations, "Crane Accidents 1997 - 1999"

Appendix D, "Crane Accidents 1997-1999," includes an Occupational Safety and Health Administration (OSHA) report concerning crane accidents from 1997 through 1999. Data for the OSHA Crane Report was gathered from Federal OSHA's Office of Management Data Services (OMDS) Website, and from Micro-to-Host reports from the Integrated Management Information System (IMIS). Unfortunately, the findings that are made in the report are gross failures, and were not normalized by load weight, crane capacity, type of industry, or the number of failures per demand.

Several observations of the OSHA report are similar to this and other crane operating reports.

- The number of crane accidents occurring during construction activities was about the same as crane accidents that occurred during non-construction activities. Of the 158 crane accidents, 80 accidents occurred during non-construction work and 78 during construction-related work. It is assumed that "non-construction" crane accidents included general routine maintenance or industrial activities involving load movements.

- Crane accidents were dominated by mobile cranes. Of the 158 crane accidents, mobile cranes accounted for 73 percent of the accidents, bridge cranes 16 percent, gantry cranes 3 percent, tower cranes 3 percent, and ship cranes 1 percent. There were 7 crane accidents (4 percent) where the type of crane involved was not known.

- More accidents occurred in the private sector. Of the 158 crane accidents, 150 accidents involved private sector entities and 8 involved public sector entities. Of the 8 public sector cases, 7 resulted in serious injuries.

- Public sector crane accidents were dominated by mobile cranes. All 8 of the public sector cases involved a mobile crane.

7.5 Navy Crane Events

Appendix E, "Navy Crane Operating Experience," provides information on U.S. Navy crane operating experience data for the period 1995-1999. Operating experience obtained from the Navy has been used by industries utilizing cranes, to reduce the risk and financial impact of crane accidents. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (published in 1980) relied heavily on U.S. Navy crane operating experience. The Navy crane data used in NUREG-0612 included summaries of 466 crane events covering a period from February 1974 to October 1977.

For the period 1995-1999, the U.S. Navy reported 66 crane events, as shown in Appendix E. Each crane event is listed by crane type, accident type, accident cause, responsible group, function being performed at the time of the event, and crane operating mode. A breakdown is also provided showing the end result of the crane event and its cause. As shown in Navy crane data for 1995-1999, human factors or human errors are the leading causes of Navy crane issues. This would include the categories of improper operation, improper rigging, and procedure failure. These three cause categories accounted for approximately 88 percent of crane issues. Those crane issues related to crane equipment failures accounted for approximately 5 percent of crane issues. During the time period, there were 11 incidents which involved loads in excess of 18 metric tons (20 tons). Four different accident types were recorded for the 11 events, (i.e., overload, damaged crane, load collision, and damaged load) most of which were caused by human errors (i.e., not following procedures or lack of skills).

An exact accounting of the number of lifts per year made by each crane was not available from the Navy. Estimates were made of the number of lifts, and of the number of load drops due to changes in the number of facilities and vessels covered in the reporting system.

8 CRANE OPERATIONAL EXPERIENCE AND INSIGHTS

The information presented in this section include findings from a review of crane operating experience at U.S. nuclear power plants and of reports available from other sources on crane operating experience. These insights can be used to initiate actions to reduce the likelihood of a load drop caused by equipment malfunctions or human errors.

8.1 The Human Error Rate for Crane Operating Events Has Significantly Increased

The percentage of crane issue reports caused by poor human performance has increased over time, and for the last several years, averaged between 70 and 80 percent. The average percentage of crane issue reports caused by poor human performance for the entire time period (1969 through 2002) was calculated to be 73 percent. “Not Following Procedures” was the largest contributor. Other categories that are similar to “Not Following Procedures” would be “Ventilation” (i.e., failure to establish the required ventilation prior to load movements in certain areas), “Did Not Test” (i.e., failure to perform crane surveillance tests prior to use) and “Load Path” (i.e., failure to move loads over established safe load path areas). Similar human error results were reported in a 1996 DOE report, “Independent Oversight Special Study of Hoisting and Rigging Incidents Within the Department of Energy [DOE].” The DOE report presents the results of an analysis of hoisting and rigging (H&R) incidents, covering the period from 1993 to 1996. Human error, whether directly associated with supervisors or equipment operators represented approximately 94 percent of DOE H&R incidents. As shown in Navy crane data for 1995-1999, human factors or human errors are the leading causes of Navy crane issues, in that, the categories of improper operation, improper rigging, and procedure failure, accounted for approximately 88 percent of Navy crane issues. Navy crane equipment failures accounted for approximately 5 percent of crane issues.

8.2 The Human Error Rate Is Lower When Lifting Very Heavy Loads

The human error rate for very heavy loads is less than the human error rate when considering all load weights. When considering only very heavy loads (e.g., loads greater than 27 metric tons [30 tons]), the percentage of crane issue reports caused by poor human performance is 56 percent v.s. 73 percent for all load weights.

8.3 Load Drop Events Have Increased in the Last Decade

During the period 1969-2002, there were 57 reported events involving load drops. Load drops while operating the spent fuel pool crane (representing over half of the load drop events) were largely because of fuel assembly drops caused by grapple operation or human errors which posed no safety issue. Load drops while operating mobile and other cranes (representing almost half of the events) have occurred outside of safety related areas. However, several load drops have involved overhead cranes similar to those used in safety-related areas of the power plant. When compared to the previous decade (1981-1992), the last decade (1993-2002) experienced a 60 percent increase in the number of load drop events, concurrent with an increase in the number of operating units by 9 percent.

8.4 Below-the-Hook Crane Events Have Greatly Increased

For the period 1968 through 2002, there were 47 reported below-the-hook events, many resulting in load drops and damaged equipment. Over the last decade (1993-2002), there were 33 below-the-hook events, of which 17 involved load drops, 10 involved equipment damage, four involved administrative issues, and two involved load slips. During this period, the number of events increased by 230 percent (when compared to the previous decade), concurrent with an increase in the number of operating units by 9 percent.

8.5 Inconsistent Load Drop Calculation Methodologies and Consequences

Calculational methodologies, assumptions, and predicted consequences varied greatly from licensee to licensee for very similar accident scenarios. Accurate load drop analysis is essential, since each licensee uses load drop calculations to determine transport height restrictions which are referenced in their heavy load lift procedures. Load drop analyses also determine locations where other measures besides load height restrictions are necessary (e.g., impact limiting devices, interlocks to prevent crane motion over certain areas, or employment of single-failure proof handling systems).

8.6 Very Heavy Load Drops at Boiling Water Reactors Are at Greater Risk Than at Pressurized Water Reactors

In general, very heavy load drops in BWR plants are more risk significant than very heavy load drops in PWRs because of plant systems layout, in that, for PWRs, spent fuel cask transfer occurs in an area separate from the reactor building and many safety related systems. However, for BWRs, many very heavy loads are commonly lifted and moved on the upper floor of the reactor building or the auxiliary building. Should a floor breach occur as the result of a load drop at a BWR, there are many safety-related components located on lower floors which could be damaged or disabled. This situation is worsened for BWRs that have a Mark I containment which places the torus directly below the equipment hatch in the reactor building. A load drop in certain areas could simultaneously initiate an accident, and disable accident mitigation equipment. These types of events have the potential to defeat defense-in-depth.

8.7 There Were No Accident Sequence Precursor Events Involving Crane Operation

There have been no Accident Sequence Precursor (ASP) events for the period 1985 through 2002 that involved a crane. To be classified as an ASP event, the event must have a conditional core damage probability (CCDP) of at least 1.0E-06. The most risk significant crane events have been those resulting in a loss of power. There have been 10 loss of power events caused by crane operation from 1968 through 2002, nine of which were caused by mobile cranes. Of the nine mobile crane events involving a loss of power, two events had Augmented Inspection Team (AIT) inspections (Palo Verde and Diablo Canyon). During the last decade (1993-2002), there were three events that resulted in a loss of power. This represents a reduction of 43 percent from the preceding decade, concurrent with an increase in the number of operating units by 9 percent.

8.8 The Number of Mobile Crane Events Has Declined Slightly

There have been 38 recorded events involving mobile crane operation from 1969 through 2002. Many of these resulted in tip overs, load drops, and equipment damage. Several mobile crane events have resulted in a loss or partial loss of power to various electrical lines servicing plant equipment. During the first decade (1969 through 1980) there were six events of which five occurred at plants under construction. Mobile crane performance progressively worsened during the second decade (1981 through 1992) when there were 17 events of which four occurred at plants under construction. During the third decade (1993 through 2002) an improving performance trend occurred with a slight reduction in the number of events when compared to the previous decade (1981 through 1992).

8.9 Radiation Exposure Events During Crane Operation Were Caused by Human Error

There were three crane events that resulted in radiation exposures, each were caused by human error. At the Pilgrim facility in 1979, a crane operator lifted an irradiated fuel assembly out of the spent fuel pool, resulting in increased exposure. At Turkey Point Unit 3 in 1992, a maintenance person was inattentive during movement of the polar crane, and fell into the refueling cavity and got contaminated. A third radiation event caused by crane operation occurred at Farley Unit 2 in 1999 when the failure of the polar crane primary height measuring system allowed a portion of the reactor lower internals to be exposed during a lift. None of the radiation events was caused by a load drop or slip, and none were significant.

8.10 The Fuel Assembly Drop or Damage Rate Caused by Crane Operation Has Decreased

There have been 30 crane events involving either a fuel assembly drop or damage to a fuel assembly caused by handling. However, given the steady increase in the number of operating units to over 100 during the period of the survey, there was an overall improvement in time in fuel handling performance. From a risk perspective, none of the 30 fuel assembly drop or fuel handling events resulted in radiation exposure or risk to personnel.

8.11 There Were Few Load Slips Involving Very Heavy Loads

Of the estimated 54000 very heavy load lifts at operating facilities following the issuance of NUREG-0612 in 1980, there were six very heavy load slips. None of the six very heavy load events resulted in radiation releases, risks to licensee personnel or the public. In 1999, Comanche Peak Unit 1 had the most significant very heavy load slip event involving the slip of a reactor coolant pump motor of 4.6 to 6.1 meters (15 to 20 feet). As the load was rapidly falling, one link of the hoist chain randomly lodged in the lower chain block which arrested the unplanned descent. The motor stopped approximately 2.4 meters (eight feet) above the pump base. Had the link not lodged in the chain block, the motor could have continued dropping, damaging the reactor coolant pump and piping. The issue was of very low safety significance in the reactor safety strategic area because all fuel had been transferred to the spent fuel pool prior to the load slip. However, at the time of the slip, load control procedures allowed performance of this very heavy load lift in operational modes 5 or 6, where fuel would be present in the reactor vessel. Damage to reactor coolant system integrity in modes 5 or 6 significantly increases the probability of fuel damage because mitigating equipment necessary to recirculate lost coolant is not required to be available.

8.12 There Were Few Load Drops Involving Very Heavy Loads

Of the estimated 54000 very heavy load lifts at operating plants since the issuance of NUREG-0612, three load drops were identified. These three very heavy load drop events occurred because of human error, and ultimately because of rigging deficiencies and not because of crane deficiencies. The three events also did not occur near any safety related areas, and none resulted in radiation releases, risks to licensee personnel, or the public. The Byron very heavy load drop event occurred while operating a mobile crane, while the San Onofre 3 and Turkey Point 4 very heavy load drop events occurred while operating turbine building overhead cranes.

8.13 Estimates of Load Handling Failure Rates Are Low

Based on actual crane operating experience data from commercial U.S. nuclear power plants, this study estimates the rate of load drops per demand for very heavy loads to be 5.6E-05. This estimate is an industry average, and may be higher or lower at a given facility because of varying human error rates which appear to dominate load drop events. NUREG-0612, which based its estimates on the data collected from the Navy, estimated the probability of a handling system failure for nuclear plant cranes will be on the order of between 1E-05 and 1.5E-04 per lift. This probability of failure was an estimate since Navy crane data does not indicate how many lifts were actually performed, i.e., only the number of problems have been quantified. A report issued by the Environmental Evaluation Group (EEG) of New Mexico, estimated the probability of failure of the TRUDOCK crane system at the Waste Isolation Pilot Plant (WIPP), to have a combined equipment failure rate per demand of 5E-06, and a combined operator error rate of 1.7E-07 per demand.

8.14 The Criteria for Single-Failure-Proof Crane Classification Has Been Inconsistently Applied

Although single-failure-proof cranes share many common design features (e.g., dual reeving, redundant limit switches, and redundant brakes), the remaining criteria for declaring a crane as single-failure-proof (e.g., for new cranes or upgraded cranes) have been inconsistently applied. Crane manufacturers have also stressed that NUREG-0554 is ambiguous in some areas, and that clarifications or changes need to be made to both NUREG-0612 and NUREG-0554. Industry suggested that a preferred approach would be to consider adopting ASME NOG-1, Type I (with minor changes) as an acceptable approach to meeting NUREG-0554 and for upgrading cranes to single-failure-proof status. NOG-1 contains much more specific design criteria for single-failure-proof cranes than does NUREG-0554. In addition, while some licensees listed a crane as single-failure-proof, or indicated that it met NUREG-0612 upgrade requirements, all the single-failure-proof design criteria listed in NUREG-0554 still may not be fully met. Among events occurring during the period 1968 through 2002 involving cranes suitable for an upgrade to a single-failure-proof design, most load drop events have been the result of poor program implementation or human performance errors that led to hoist wire rope or below-the-hook failures. All three very heavy load drops were the result of rigging failures, not crane failures. Consequently, there were no very heavy load drop events that could have been prevented had only a single-failure-proof crane been employed in the lift. However, there were load or hook and block assembly drops that could have been prevented with the use of single-failure-proof cranes and lifting devices.

8.15 Many Generic Communications Were Issued by the NRC Involving Crane Operation

The accuracy and consistency of the information received by the NRC is questionable. There have been 29 NRC generic communications (GCs) that have involved load movements at U.S. nuclear power plants dating back to 1976. There have been nine GCs which discussed heavy loads moved on the refueling floor, load drop analysis for heavy loads, identification of heavy loads that are lifted over safe shutdown equipment, and the consequence of a load drop on selected equipment. A few GCs (issued as generic letters and one bulletin) requested licensees to provide information on their crane programs for NRC evaluation. Many of the licensees that responded to the latest request (Bulletin 96-02), provided incomplete information. Also, in many instances, information previously provided to the NRC was not verified to be accurate.

8.16 Few Licensees Have Performed a Consequence Analysis for Heavy Load Drops

Although not required by NRC regulations, few licensees have performed a consequence analysis of heavy load drops. Of the 74 facilities that responded to Bulletin 96-02 which requested licensees to provide the NRC with specific information relating to their heavy load programs and plans, eight licensees indicated that a consequence analysis had been done at their facility for heavy load drops.

8.17 Injuries Caused by Crane Operation Has Increased in the Last Decade

The number of crane-related injuries has increased during the last decade. There have been 16 reported injuries involving crane operation during the period from 1969 through 2002. When comparing the last decade (1993-2002) with the second decade (1981-1992), a 100 percent increase in the number of injuries occurred concurrently with a 9 percent increase in the number of operating power plants.

8.18 Deaths Caused by Crane Operation Occurred Largely During Construction

There have been 10 reported crane events that have led to deaths in the nuclear industry for the period 1969 through 2002. The highest concentration of crane related deaths at nuclear power plants occurred during the first decade (1969 to 1980). For the first decade, six of eight events that led to a death occurred at facilities still under construction. The last death in a crane related accident in the U.S. nuclear industry was 1985.

9 REFERENCES

1. U.S. Nuclear Regulatory Commission, memorandum, “Proposed Generic Safety Issue - Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,” April 19, 1999.

2. U.S. Nuclear Regulatory Commission, memorandum, “Proposed Generic Safety Issue - Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,” May 27, 1999.

3. U.S. Nuclear Regulatory Commission, NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” May 1979.

4. U.S. Nuclear Regulatory Commission, NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants, “ July 1980.

Appendix A

Crane Events at U.S. Nuclear Power Plants 1968 through 2002

Introduction

A review of crane documents in the NRC's Nuclear Document System (NUDOCS), the NRC's Agencywide Documents Access and Management System (ADAMS), events reported by individual licensees, through NRC documents and inspection reports, by vendors, and the public for the period 1968 through 2002 resulted in 431 different issues. Depending on the severity of each issue, each issue may be discussed in several documents. Most crane issues are administrative (not following a procedure, load path issues, noncompliance with technical specifications, inadequate crane operational testing prior to use, etc.). A few crane issues relate to problems encountered when lifting loads of approximately 27 metric tons (30 tons) or more. The data and resultant sorting is shown on Table A1, "Reported crane issues at U.S. nuclear power plants." Abbreviations for nuclear power plants are shown on Table A2, "Plant name abbreviations."

Sorting of Crane Issues

To analyze crane issues, six general categories were established, most with several subcategories. Once this information was entered in the database, sorts were performed to look for trends and patterns.

- Category 1: Plant and event report date

  Subcategories include; plant docket number, plant name, event report year, event report month, whether the crane issue occurred when the plant had an operating license and also occurred after January 1980 (Post NUREG-0612), date of commercial operation, and the shutdown date.

- Category 2: Crane type

  Subcategories include; reactor building, polar, auxiliary building, refueling/manipulator, spent fuel pool, tower, mobile, and other.

- Category 3: Crane deficiency

  Subcategories include; structures, control systems, brakes, rails, fasteners, below-the-hook, unknown or indeterminate, and none.

- Category 4: Reported administrative cause for event

  Subcategories include; not following procedures, poor procedures, failed to test, load path inadequacy, ventilation inadequacy, maintenance, engineering, operations, unknown or indeterminate, and none.

- Category 5: Safety implication of event

  Subcategories include; Death, injury, radiation release (RAD), load slip, load drop, very heavy load, crane component drop (above the hook), equipment deficiency or damage, loss or partial loss of power, fuel drop or damage, and none.

- Category 6: Load description for slip or drop events

  Subcategories include; Issue abstract description, and drop or slip distance.

Table A1: Reported crane issues at U.S. nuclear power plants

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
IP2	68							Multiple cracks were found in the polar crane rail.	
GIN	69					YES		The core barrel, thermal shield, lower core plate and attached internals weighing approximately 82 metric tons (90 tons) dropped to their stand following a brake failure.	1.8 meters (6 feet)
IP2	69							While lowering the lower core internals weighing approximately 136 metric tons (150 tons) into the reactor vessel, one phase of the electrical supply was lost. The load lift was ceased. No damage was done.	
YR	69							During refueling, a fuel bundle was damaged. The assembly retainer band on the carriage had been crushed against the fuel assembly lower nozzle. In addition, a welded joint on the upender was cracked. The event was caused by operator error.	
PAL	70					YES		A 23 metric ton (25-ton) capacity auxiliary hoist on a polar crane two-blocked when the operator bypassed the interlocks, parting the cable, resulting in the CRDM support tube, hoist sheave, and hook to fall (0.95 metric ton [1.05 tons).	6.7 to 7.9 meters (22 to 26 feet)
TP3	70	YES	YES			YES		A special crane erected on the turbine pedestal collapsed when two vertical support cables snapped while lifting the Unit 3 generator stator into its permanent location, killing one person and injuring two others.	61 cm (2 feet)

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
IP2	71							A vertical pedestal leg of the polar crane was damaged (structural section not vented, resulting in section becoming dented in) during performance of a containment pressure test.	
IP3	71					YES		While lowering the pressure vessel weighing 402 metric tons (443 tons) the cable parted and gear bracket welds failed. Only minor damage was done to the vessel.	Short
FER1	72					YES		While moving a fuel assembly from the fuel storage facility to the fuel and repair building, the crane was two-blocked, damaging a shackle, resulting in the fuel assembly falling into the transfer tank.	8.2 meters (27 feet)
HN	73	YES						An employee died after falling 3 meters (10 feet) from an overhead yard crane.	
VY	73							A grapple came lose from the jib crane hoist cable.	
MIL1	74					YES		A fuel assembly became detached from the grapple and fell in the spent fuel pool.	Not specified
PIL	74					YES		An irradiated fuel assembly became detached from the grapple and fell in the spent fuel pool.	Not specified
DA	75					YES		A fuel assembly became detached from the grapple and fell in the spent fuel pool.	Not Specified
HUM	75					YES		A fuel assembly became detached from the grapple and fell in the spent fuel pool.	Not specified
IP3	75							The crane used to remove and relocate fuel elements sustained malfunctions which in turn damaged the fuel-handling equipment.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
RS	75							Incorrect fasteners, loose bolting, and a missing seismic support discovered on the polar crane during an inspection.	
SAL12	75							Crane hooks were replaced after the discovery of multiple indications.	
SUR1	75							During movement of a manipulator crane, a control rod drive shaft was bent when the manipulator outer mast was inadvertently driven into the upper internals package, damaging the drive shaft.	
SUR2	75							A manipulator crane stopped due to load overload. Perforations in the fuel were discovered.	
BRU2	76					YES		A fuel assembly became detached from the grapple and fell in the spent fuel pool. The assembly fell to a horizontal position across the top of the spent fuel pool storage racks.	Not Specified
BRU2	76					YES		A fuel assembly became detached from the grapple and fell in the spent fuel pool. The assembly fell before it was fully inserted into its rack.	Not Specified
CP12	76	YES				YES		While lifting a personnel bucket (unoccupied) with a mobile crane, it became unbalanced. The crane boom failed, coming to rest in the turbine mat area.	Unknown
DRE23	76				YES			While lowering the reactor pressure vessel head to reinstall it, the head dropped abruptly 38 cm (15 inches) before the brake engaged. A second abrupt drop was observed before the head was seated on the reactor vessel flange. The slips were apparently caused by a modification which added an inching motor to drive the hoist at slower speeds.	38 cm (15 inches)

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
OCO23	76							Polar cranes in units 2 and 3 were operated without an approved procedure.	
PEB23	76	YES						A worker died after falling 15.3 meters (50 feet) from the radwaste building crane hook.	
PTB12	76		YES					An operator was injured when a gantry crane came off its track.	
RS	76							A loaded polar crane traveled over an open reactor vessel.	
RS	76							A lift was made using the polar crane that was not allowed by the technical specifications.	
HN	77							An overhead crane inadvertently lifted up the spent fuel pool rack 31 cm (12 inches) and again 15 cm (6 inches). The lift was caused by a malfunctioning switch. The racks were not damaged.	
MON	77							The reactor building crane was modified to include a new trolley. The crane was not adequately tested following the modification.	
OC	77				YES			A fuel assembly and mast dropped while lowering the assembly into the spent fuel pool racks. The drop was arrested by the cable drum brake. However, the slip resulted in shearing six bolts that coupled the refueling mast speed reducer to the cable drum. An examination indicated that 4 of the 6 bolts had failed at some earlier date.	
PEB3	77					YES		A fuel assembly was inadvertently released from the grapple and fell across the core. The cause was attributed to operation of the grapple open switch on operator error.	Not specified

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BYR12	78							The turbine building crane cable broke when over stressed. This was caused by a mismatch of voltage balancing resistors in the circuit board (10CFR Part 21 report).	
CAL	78		YES					Two workers were injured when a support girder for the polar crane rail fell on them. Rigging caught on the girder causing it to fall.	
CRY	78					YES		A missile shield crane hook failed, dropping the new fuel elevator test weight on a fuel assembly causing minor damage. The crane hook was plant fabricated.	Short
GG	78							A tornado toppled the Unit 1 north and south tower cranes. The containment building was hit by the crane.	
NMP2	78	YES	YES					A bundle of rebar was hit by a crane carrying a load, resulted in two deaths and eight injuries.	
SHO	78							Approximately 10 percent of the polar crane welds were found to be defective.	
TMI2	78							The load reading on load lift cell exceeded the procedure maximum while lifting the reactor vessel head.	
BELL	79							A 227 metric ton (250-ton) capacity Link-Belt mobile crane collapsed into the cooling tower due to high winds. Damage to the cooling tower was minor.	
DCC2	79							Surveillance tests on the crane were not completed prior to moving fuel.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
DRES2	79							A surveillance test on the crane was not performed prior to using the reactor building crane.	
PER	79	YES						A worker was killed when he touched a mobile crane that was in contact with an overhead high voltage electrical line.	
PIL	79			YES				The crane lifted irradiated fuel assembly out of the spent fuel pool resulting in increased exposure.	
PIL	79					YES		A new fuel assembly was being moved to the spent fuel pool using the reactor building crane, when the assembly struck the top edge of the high density fuel racks and the latching device on the auxiliary hook failed to retain the fuel. The assembly fell, striking the lifting balls on four spent fuel elements, then coming to rest on the top of the fuel racks.	A few meters (several feet)
RS	79							Loose bolting was discovered on the polar crane.	
SAL1	79							A total of 31 fuel assemblies that were removed from the core had suffered some grid damage due to load movements.	
SH12	79		YES					A mobile crane overturned into the Unit 2 reactor auxiliary building and injured three workers. Some damage was done to the auxiliary building.	
BYR2	80	YES						A contractor died after being caught between the tower crane counter weight and the engine housing.	
DA	80							Fuel movement in the core was performed with the control rods not fully inserted.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
HART	80		YES			YES		While pouring concrete, the crane brake failed dumping concrete and severely injuring three workers.	Short
MH12	80	YES						A mobile crane got stuck in the mud while lifting a load. The crane tipped over, killing the operator.	Short
MH2	80							A collision of the ringer crane and the Unit 2 containment liner caused minor damage.	
MILL3	80							Bolts broke during assembly of the truck girder for the polar crane.	
PER	80							A high failure rate of polar crane girder welds was reported.	
RS	80							The polar crane traveled over the upender that was loaded with irradiated fuel.	
SAL1	80							The manipulator crane was not properly load tested prior to use.	
SH12	80		YES			YES		A crane inside containment lifted a lifting tackle approximately 43 meters (140 feet) when the binding broke. The lifting tackle fell, landed first on scaffolding, and then onto eight workers. Various injuries were received. The cause is unknown.	43 meters (140 feet)
STP12	80							Several broken tie down studs were found on the polar crane rails. In addition, the curvature of the rails failed to meet design specifications.	
STP12	80							Multiple fastener and structural issues were identified with the polar crane rails, including design and installation.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
TMI12	80					YES		A mobile crane tipped over while moving a load of scrap metal outside the gate.	Short
VOG1	80							A tower crane collapsed inside the 1B cooling tower. One person was injured. Other damage was not specified.	
WC	80							A polar crane rail was found broken. The extent of the failure was not specified.	
WC	80		YES					A contractor was seriously injured when a crane boom fell while it was being dismantled.	
WNP3	80							A tower crane collapsed into and behind the WNP-3 reactor auxiliary building. The extent of damage was not specified.	153 meters (500 feet) tall Tower crane
CAL	81					YES		A crane boom collapsed on service water building while lifting a concrete hatch weighing 11 to 16 metric tons (12 to 18 tons).	Unknown
DCC1	81					YES		A fuel bundle was damaged while it was being transferred in the refueling cavity using the manipulator crane, when the lower end of the assembly struck a ledge on the refueling cavity floor just outside the reactor vessel area. One rod was dislodged and fell from the assembly onto the refueling cavity floor. No radiation was released.	
DRE23	81							The reactor building crane girder was damaged when a lifting fixture was raised too high, and impacted the box girder, leaving dents that were about 2 and 3/4 inches deep.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
OCO3	81							The polar crane was parked over the fuel transfer canal with the vessel head removed.	
PI1	81					YES		The top nozzle and fuel assembly weighing less than 0.9 metric ton (2000 pounds) separated causing the assembly to drop. The failure was caused by IGSCC.	Not specified
RB	81							A Manitowoc 4600 W mobile crane tipped forward, causing the boom to strike the standby cooling tower basin which was under construction. Damage was restricted to rebar and not concrete.	Unknown
SUS1	81							Tests on the crane were not completed prior to its use.	
WNP14	81							Inadequate load restrictions for the fuel cask handling crane were noticed.	
DCC1	82							The upender device had not been raised to the vertical position before the fuel assembly was lowered. This resulted in the fuel assembly becoming cocked and lodged in the manipulator mast. Minor deformation marks and scratches were noticed on a few rods. There was no radiation release.	
MH2	82							Three cracked welds were discovered in the polar crane structure.	
PER	82							The Unit 2 polar crane girders had defective welds.	
RS	82							Fifteen broken bolts were found on the polar crane hold down clips due to fatigue.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BRU1	83							Fuel movement in the core was performed with control rods fully withdrawn.	
DRE3	83							While lifting a load, a mobile crane boom contacted a drain valve on a sprinkler system causing a break.	
FER2	83							The reactor building crane hoist stalled, caused by faulty hoist motor wiring of the thermal overload circuits.	
MILL2	83							A spent fuel pool gate was lifted over irradiated fuel.	
PER	83							While attempting to remove the shroud head/separator (weighing approximately 50 tons) from the reactor pressure vessel, the strongback lifting device was broken, because the securing fasteners were not removed prior to the lift.	
RB	83					YES		The reactor shield building dome form assembly weighing approximately 363 metric tons (400 tons) fell following the buckling of a Lampson Traslift crane mast.	9 meters (30 feet)
TMI12	83						YES	A mobile crane at TMI-2 made contact with a power line (230 kV) which was a source of offsite power for both units. It resulted in the loss of one of two trains of safety related electrical distribution busses in both units.	
TP4	83					YES		A fuel assembly was being inserted into the core. It was not aligned properly, and fell over so that it leaned at a 35 degree angle against two other fuel assemblies.	Tip

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
TP4	83					YES		During a fuel bundle lift from its storage rack, the limit switches failed to stop upward movement. The hoist two blocked, parting the hoist cable and causing the assembly to drop back into its rack .	Not specified
BF1	84							The spent fuel pool crane limit switches were out of adjustment, allowing the crane to travel in restricted areas.	
BRU1	84							Crane was inoperable due to damaged electrical leads.	
CRY	84							Load movements were not terminated when radiation counts in the area exceeded their threshold.	
DB	84							Fuel movement was made without establishing adequate ventilation.	
DB	84							The polar crane was loaded with a plenum assembly lifting rig weighing approximately 8.4 metric tons (9.3 tons) with no operator present.	
DCC1	84							Fuel movement occurred without establishing adequate ventilation.	
FER2	84							The hoist motor on the reactor building crane failed due to maintenance personnel installing motor leads out of phase. The hoist was not properly tested following completion of the maintenance work.	
FTC	84							A load weighing 113 kgs (250 pounds) was carried over the reactor coolant system when the system temperature was greater than 107 degrees Celsius (225 degrees Fahrenheit).	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
HAT1	84					YES		A possible inadvertent actuation of the fuel grapple hook position switch resulted in dropping a spent fuel bundle about 3.7 meters (12 feet) into its storage rack cell, slightly deforming and scratching the bundle and rack. No radiation release occurred.	3.7 meters (12 feet)
LAS2	84							Surveillance tests were not performed to verify that circuits were de-energized.	
MILL2	84					YES		A spent fuel pin was dropped while performing eddy current testing. The cause was attributed to an inadequate gripping force on the pin.	Not specified
OC	84							Spent fuel pool gates were lifted over fuel.	
OC	84							Lift height limit switches were not properly calibrated and were over ridden by the crane operator when lifting a spent fuel shipping cask over the spent fuel pool cask drop protection system.	
PAL	84							While reloading the core, a new fuel bundle stuck in the refueling machine. A low air pressure supply pressure in combination with leakage, prevented movement of the bridge trolley.	
RS	84							Slings used to lift the refueling rack failed due to excessive load, caused in part by a load cell that was set too high, and by improper rigging. The rack was safely lowered without dropping.	
SHO	84							Crane vibrations occurred during fuel movement in the spent fuel pool. Movement was ceased.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
SHO	84							A faulty fuel handling bridge crane solenoid was discovered during fuel movement.	
SON2	84							The polar crane malfunctioned during a reactor vessel head movement (resulting in a downward motion) caused by the malfunction of a power supply card.	
SUM	84							Maintenance failed to calibrate the spent fuel bridge crane load cell prior to crane use.	
TMI2	84							While lowering shielding over the reactor internals, the crane stopped lowering.	
BF2	85	YES	YES					A 23 metric ton (25-ton) capacity turbine building crane hook fell thru a temporary building in the turbine building resulting in one death and three injuries.	Several meters (Many feet)
DCC1	85							The auxiliary building crane lifted a load over the spent fuel pool.	
HAT1	85							The turbine building crane hook collided with a train "A" deluge pressure gage, resulting in a flood. Subsequently, clogged drains resulted in water intrusion into the control room.	
MCG2	85	YES						A crane operator died after trying to step onto the manipulator crane, but fell back and became lodged between the crane and an electrical panel.	
PER	85							Upon inspection, it was found that the polar crane box girder had several defective welds.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
STL1	85				YES			While performing a lift of the upper guide structure weighing approximately 45 metric tons (50 tons), a bolt used to help secure a rigging device failed because it was improperly threaded. The upper guide structure tilted approximately 15 cm (6 inches) when the bolt failed. No damage was done.	
TMI1	85							Polar crane operating procedures were not followed.	
TMI2	85					YES		A defueling canister and support sleeve weighing approximately 1 metric ton (1.1 tons) fell into the reactor vessel when is was dislodged from the positioning system using a jib crane. In addition, the jib crane was rated at 0.9 metric ton (1 ton), while the load was 1 metric ton (1.1 tons).	46 cm (18 inches)
ZIO1	85							A load was moved that was heavier than allowed. The load traveled over the spent fuel pool. In addition, the interlocks were inoperable.	
CAT1	86							The auxiliary building crane was used to remove two control rod drive assemblies, rather than the manipulator crane.	
CP12	86							The polar crane rail clips were welded, caused by a poor engineering modification.	
DC1	86							A fuel assembly was damaged during refueling when it was not properly aligned with the pins in the lower core support plate. Some damage was done to the assembly grid strap but not to the fuel.	
FAR1	86							Two control rod drive assemblies were removed from the reactor cavity using the wrong procedure and the wrong crane.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
HN	86					YES		During the lift of the upper core support structure weighing approximately 26 metric tons (28.5 tons), a fuel assembly stuck to the structure because of a bent fuel assembly locating pin. The assembly fell off when the load was moved laterally. The dropped assembly and the two fuel bundles that it impacted were damaged, but there was no radiation release.	61 to 122 cm (2 to 4 feet)
MCG1	86							Loads were moved over the spent fuel pool without establishing adequate ventilation.	
PAL	86							Eight lifts of missile shields were performed assuming that they each weighed 32 metric tons (35 tons), when they actually weighed 58 metric tons (64 tons) each. The lifting device used to lift the missile shields had a safe working load of 47 metric tons (52 tons).	
SAL2	86							Maintenance personnel failed to test an overload cell prior to crane use.	
TMI2	86							A polar crane brake modification to over-ride brake controls was determined to be a violation.	
WC	86							The spent fuel pool overload cell was not tested prior to use, and the crane lifted a load that was greater than the maximum allowed.	
CAL	87							The cask handling crane was not tested before being used.	
CAL	87							The spent fuel pool crane was not tested prior to being used.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
CP12	87							The configuration of the refueling crane rails for use with the roll-away missile shields were found not to be seismically qualified.	
CP12	87		YES					A mobile crane tipped over injuring 10, causing property damage. The event was caused in part by high winds.	
CP12	87							There were an insufficient number of polar crane rail clips installed.	
CRY	87						YES	A small crane hook collided with a breaker cubicle, shorting two phases and causing an under voltage. This caused the "A" engineered safeguards train to be inoperable, while the "B" train was out of service.	
DCC1	87							A refueling manipulator crane load cell calibration problem.	
DCC1	87							An auxiliary building crane traveled over the spent fuel pool.	
GG	87							The reactor vessel head was lifted over the spent fuel pool.	
GG	87					YES		A container of two new fuel bundles fell off a transfer cart to the turbine deck because of crane operator and rigging issues. Both fuel bundles had minor damage and were not used.	61 cm (2 feet)
HN	87							During a maintenance activity, the spent fuel pool crane experienced a partial loss of non-vital 480 V power. During the power loss, a fuel assembly was suspended by the crane above the spent fuel racks.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
PEB2	87						YES	A mobile crane caused a trip of a transmission line when it drew an arc from a 220kV electrical line, causing a partial loss of offsite power.	
PV1	87							The new fuel handling crane was not tested prior to use.	
SAL12	87							A mobile crane collided with a fire protection valve creating a significant leak.	
SAL12	87							The fuel handling crane was not tested prior to use.	
SAL2	87							A fuel handling crane made two lifts over the spent fuel pool without first being tested.	
SAL2	87							Load tests were not completed prior to crane use.	
TMI2	87							The maximum load lift height was exceeded by more than 1.2 meters (4 feet). The wrong procedure was also used for the lift.	
TP4	87							Corrosion damage was reported to be done to spent fuel pool rails, hold down clips, and fasteners.	
FTC	88							A load was carried over the reactor coolant system when the coolant temperature exceeded 107 degrees Celsius (225 degrees Fahrenheit).	
MCG2	88							A heavy load was lifted over the spent fuel pool.	
PAL	88							While lifting the upper guide structure, a fuel assembly was found to be attached to the bottom. The assembly was removed without damage. The fuel bundle was stuck because of the guide pins.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
SAL2	88							Fuel movement was performed without establishing adequate ventilation.	
TMI2	88							The load maximum lift height was exceeded by approximately 15 cm (6 inches).	
CC1	89							The spent fuel cask crane block was lifted over the spent fuel pool.	
COL	89							The mobile crane load path included going over safety related structures to retrieve a turbine exhaust fan in the rad waste building.	
MILL3	89							During the lift of the refueling pit seal ring, a fire lasting longer than 10 minutes started in the resistor bank cabinet for the polar crane auxiliary hoist. The seal ring was suspended approximately 1.5 meters (5 feet) below the reactor top hat until repairs were made.	
MILL3	89							Fuel movement was performed without establishing adequate ventilation.	
OC	89							The spent fuel pool gate was lifted over irradiated fuel.	
QC1	89							Operators lowered the reactor building crane hook until it contacted a new fuel bundle stored on the refueling floor. Also, a signal man was not present.	
QC1	89					YES		While moving a new fuel bundle, it became detached from its grapple and fell onto fuel in the spent fuel pool.	Short
SHO	89							Fuel movement was performed without establishing adequate ventilation.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
SUR12	89							A mobile crane tipped over and produced an oil spill. Additional information was not available.	
BYR2	90							The upper internals were lowered over the upper internals storage stand bending eight guide pins. The event was caused by the crane floor director who gave the signal to lower the upper internals prematurely.	
BYR2	90					YES		A fuel assembly slipped out of the basket and dropped to the top of an empty fuel rack.	Short
DCC2	90							Polar crane trolley fasteners were found to be inadequate.	
FTC	90				YES			While lowering the reactor head, the load shifted, resulting in bending two alignment pins, and scratching the head flange.	Short
IP3	90					YES		While lifting the upper core support structure weighing approximately 54 metric tons (60 tons), two fuel assemblies were found to be attached. One of the assemblies dropped into a retrieval basket when the brakes on the overhead crane were applied. A guide pin on each assembly was bent. The guide pins were most likely damaged during the previous refueling outage.	Not specified
MCG1	90							Fuel movement was performed without establishing adequate ventilation.	
MILL3	90							A crane hook collided with a drain valve on an oil gear box reservoir, resulting in an oil spill.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
NA1	90					YES		A fuel rod slipped out of the handling tool and dropped into its storage location due to a gripper mechanism failure.	A few meters (Many feet)
NA2	90							Fuel movement was performed without establishing adequate ventilation.	
OCO12	90							The polar crane was operated over the fuel transfer canal during fuel movement. In addition, there was no flagman.	
OCO3	90							A polar crane was operated over the fuel transfer canal during fuel movement.	
PV3	90							A 9.1 metric ton (10 ton) crane was moved over the spent fuel pool.	
SON1	90							Design of the brake drum for the hoist was inadequate.	
TMI12	90							It was discovered that approximately 10 percent of crane inspections were not performed.	
TMI2	90							A polar crane was moved to an exclusion zone with the main crane hoist energized.	
WC	90							The spent fuel crane was moved while the handling tool was still connected to the fuel assembly in the test location.	
CP1	91							A surveillance test was not performed prior to using the crane.	
DC1	91							All three EDGs were inoperable while heavy loads were moved over the spent fuel pool.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
DC1	91						YES	A mobile crane shorted the "A" phase main transformer to ground (500 kV line). The remaining electrical line, a 230 kV line was out for maintenance. The 230 kV line was restored in about 5 hours.	
DC1	91							Fuel movement was performed without establishing adequate ventilation.	
FER2	91						YES	A mobile crane contacted a 120kV overhead line. After the first contact, the driver of the mobile crane backed into the electrical line a second time.	
IP2	91							Many (47 of 140) polar crane rail anchor bolts failed because of standing water from in leakage.	
IP3	91							Many polar crane trolley fasteners were found to be inadequate.	
PV3	91						YES	A 32 metric ton (35-ton) mobile crane boom contacted a 13.8 kV line causing a partial loss of offsite power. The crane was not grounded as required by procedure. Fault current through the crane resulted in small asphalt fires where the outrigger pads made ground contact.	
SAL12	91							A fuel line for a mobile crane broke, spilling 76 liters (20 gallons) of fuel.	
SEA	91							Fasteners on the polar crane trolley were found to be inadequate.	
STP1	91							Fuel movement was performed without establishing adequate ventilation.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
WC	91							A crane was operated over the spent fuel pool while both EDGs were inoperable.	
CP1	92							The polar crane was operated in the area of the vessel head, contrary to procedure.	
DCC1	92							Fuel movement was performed without establishing adequate ventilation.	
FSV	92							Fuel movement was performed without establishing adequate ventilation.	
HC	92							A mobile crane hydraulic line failure resulted in 10 gallon leak.	
NA12	92							Inadequate crane testing for heavy load lifts was discovered.	
NMP2	92						YES	While placing concrete, a partial loss of offsite power was caused by the boom of a mobile crane. One of two 115kV lines lost power. The plant impact was a loss of offsite power to the Division I and III emergency buses.	
SON2	92							A mobile crane was parked to close to the Unit 2 auxiliary transformer.	
STP12	92							The slow speed controller for the polar crane malfunctioned, requiring jogging the high speed control to move loads.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
STP12	92							The manipulator crane experienced multiple problems during the 1992 refueling outage. Lift switches malfunctioned, erratic motor drive brake performance, a worn motor coupling, excessive heat buildup, and the lower mast jammed in the stationary mast.	
TP3	92			YES				During polar crane operation, a maintenance person fell into the refueling cavity, and got contaminated.	
WC	92							Fuel movement was performed without establishing adequate ventilation.	
ANO1	93				YES			While resuming a vertical lift of the reactor head, the head lowered instead of raising.	
BF1	93							Surveillance tests were not performed prior to crane use.	
BV1	93							The Unit 1 refueling crane was driven into its stops at slow speed, breaking its drive train.	
CC1	93							The crane was not adequately tested prior to use.	
CC12	93							A two-block of the auxiliary hoist on the turbine building overhead crane resulted in the cable breaking. The hook and block assembly fell approximately 12 meters (40 feet), hitting a section of reheat cross-over piping, a gang box, and then landed on the turbine deck, damaging the grating and concrete.	12 meters (40 feet)
COL	93							Crane interlocks and setpoints were not adequately tested prior to crane use.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
DC1	93							Fuel movement was performed without establishing adequate ventilation.	
DC1	93							Fuel movement was performed without establishing adequate ventilation.	
FSV	93							Reactor building crane became detached from fuel handling machine for about an hour. The fuel handling machine could tip over given a seismic event.	
FSV	93							The reactor building crane was greatly overloaded during a lift. The reactor building crane was rated at 154 metric tons (170 tons), however the load (block of concrete from the top head) weighed approximately 218 metric tons (240 tons).	
FTC	93					YES		A mobile crane tipped over during a lift of ice deflectors. The crane fell onto and toppled a security camera tower.	Short
LIM12	93							Spent fuel casks were moved without establishing containment integrity.	
NMP2	93							A blade guide was being moved from the core into the spent fuel pool when it was released from the grapple. The operator then moved the crane and noticed that the blade guide had never been released. The operator then tried to move a fuel assembly, and discovered that the mast was bent.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
OCO3	93							An empty dry storage cask was placed in the Unit 3 spent fuel pool and was mispositioned on the cask pit stand. This resulted in the cask leaning to one side, which caused the lifting hook to partially slip off the cask trunnion during a lift attempt.	
PAL	93							During movement of a cask from the spent fuel pool, electrical cabling overheated in the control circuits for the overhead crane. When the basket and transfer cask were about 6.1 meters (20 feet) from the bottom of the pool, the resistor bank was glowing, and smoke was coming from the cabling.	
PEB23	93					YES		An empty component shipping liner weighing approximately 386 kgs (850 pounds) became disconnected because of rigging issues, and fell into the spent fuel pool cask storage area. No damage was done.	6.1 meters (20 feet)
PER	93							A radwaste crane had override switches that were held in override using adjustable wrenches. The override switches (for speed control) that were held in the override position, included the hoist override and the bridge and trolley override switches.	
PTB12	93							While loading fuel into the reactor, the air supply hose to the manipulator crane slipped off, causing the latch to fail.	
SEQ1	93					YES		During fuel loading, a bundle was inappropriately unlatched, failed to insert, and tilted against the core baffle plate at an angle of approximately 18 degrees from vertical.	Short

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
SHO	93							Heavy loads were moved in the vicinity of the spent fuel pool with an incorrect lifting attachment.	
SHO	93		1			YES		A refueling jib crane weighing approximately 4.5 metric tons (5 tons) fell from the polar crane auxiliary hook to the refueling floor when the nonredundant lifting eye broke. To balance the load the licensee attached a plasma arc welding machine to the lifting device. Minor injuries were received by a worker.	Short
SHO	93							Polar crane load paths were found to be inadequate.	
SUS1	93				YES			While lowering a fuel assembly into the core, one of the sections of the fuel handling mast dropped 25 to 38 cm (10 to 15 inches). It was determined that the mast was damaged earlier during a collision.	25 to 38 cm (10 to 15 inches)
SUS2	93							While transferring a double blade guide to the spent fuel pool, the blade guide hit the side of the reactor vessel because it was not raised high enough to clear the vessel. The following day, it was discovered that the mast was bent.	
VY	93					YES		When removing a fuel assembly from the reactor core, the assembly became detached from the grapple. The fuel assembly fell back into its original location in the core. Proper grapple engagement was not verified prior to movement.	Short

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
VY	93							A fuel assembly was being moved to a sipping can when the operator inadvertently lowered the assembly rather than raising it, causing it to strike another core component. Multiple human factors concerns were noted.	
WC	93							An inadequate spent fuel pool bridge operating procedure was discovered.	
ANO12	94							Fuel assemblies were put in the wrong location.	
COL	94							A crane was not docked in its safe storage location.	
DB	94							The crane was operated without establishing adequate ventilation.	
DCC1	94							The crane was operated without establishing adequate ventilation.	
FER2	94							A fuel bundle was mispositioned in the spent fuel pool, and then relocated without following the procedure.	
FER2	94					YES		A 32 metric ton (35-ton) mobile crane tipped over on its side when lifting a steel resin liner weighing approximately 8.6 metric tons (9.5 tons) to a transport truck. The boom struck the liner as the crane tipped over, partially crushing the top and bottom of the liner.	Short
FTC	94							The crane load path went over irradiated fuel.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
HAT1	94					YES		Seven shroud bolts were being lifted from the spent fuel pool. When the bolts were about one foot above the water, the rigging failed, and the bolt fell back into the pool, puncturing the stainless steel liner. Water from the pool drained into the area between the liner and the outer concrete wall causing the water level to drop about three inches. The rigging was not correctly fabricated.	About 12 meters (40 feet)
OC	94							Casting defects were discovered in the motor to gear box flex couplings.	
PAL	94							The crane stopped while moving a cask.	
PAL	94							The crane was operated without establishing adequate ventilation.	
QC1	94							While lowering a fuel assembly, it was improperly inserted, causing damage to the assembly and the fuel handling crane.	
ROB	94							The fuel cask crane was stored in a position over the spent fuel pool.	
ROB	94							The fuel cask handling crane limit switches were not tested.	
SEA	94							The crane was operated without establishing adequate ventilation.	
SUM	94							The licensee discovered a potential over stressing of bolted connections in the under-hung sheave nest area and support mechanism. Whiting crane indicated that 11 nuclear plants were affected.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
WAT	94							An unknown object was found to be hanging from the fuel handling machine in the spent fuel pool. The object was determined to be a fuel rod encapsulation tube that stuck onto the fuel handling tool. The licensee could not determine how it happened.	
BF3	95							The reactor building crane was not secured during a tornado watch.	
BW1	95							Crane movement was performed over the spent fuel pool.	
COO	95							Heavy load lifts were performed inside containment without maintaining containment integrity.	
DC1	95							Loads were moved over the spent fuel pool without emergency power to fuel handling building ventilation.	
DCC1	95							Fuel was moved without establishing adequate ventilation.	
FTC	95							A polar crane lift of the reactor vessel hold down ring was performed with the containment air lock open.	
IP3	95						YES	A mobile crane shorted the "C" phase of the 138kV feeder to ground while loading material into a flatbed truck.	
MILL2	95							Fuel was moved without establishing adequate ventilation	
MILL2	95							The crane was operated without establishing adequate ventilation.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
MILL3	95							Fuel was lifted higher than the maximum allowable drop height.	
PAL	95							A mobile crane collided with an over head support structure.	
PAL	95							A mobile crane severed an overhead 220 VAC power line supplying lighting to the main parking lot and guard house.	
PI12	95							There was an actuation of crane overload during a cask lift.	
SAL12	95							Fuel was moved without establishing adequate ventilation.	
SEQ12	95							Crane interlocks and mechanical stops were defeated.	
TP34	95							The gantry crane surveillance procedure was found to be inadequate.	
TRO	95							The polar crane rail was misaligned and eventually failed.	
TRO	95							A polar crane rail was found to be out of alignment. In addition, the hold down holes were flame cut.	
ZIO1	95							Fuel movement was performed without an operable radiation monitor.	
ANO1	96							A spent fuel pool gate weighing more than 2000 pounds was lifted over the spent fuel pool.	
ANO1	96							Spent fuel was moved prior to completing a charcoal sample test and analysis.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BV2	96							The load path for the removal of the RHR pump motor was near unisolable incore thimble tubes.	
CAL	96							A sling failed as a load was lifted (reactor coolant pump motor). The lift was performed outside the power block.	
DB	96							A reactor vessel lifting device was lifted over an open reactor vessel.	
DC1	96							Fuel was moved without establishing adequate ventilation.	
FTC	96							Containment integrity was breached during refueling operations.	
HN	96							Fuel was moved without establishing adequate ventilation.	
IP2	96					YES		A metal transportation container weighing approximately 2.3 metric tons (2.5 tons) was dropped when slings were set at too acute an angle. The slings slipped off crane hook.	Short
MIL1	96							A spent fuel pool gate was suspended over the spent fuel pool.	
MILL12	96							An unanalyzed load path for moving spent fuel pool gates outside of the pool was discovered.	
MILL2	96							The laydown area for the low pressure turbine hoods was directly above the safety related 480 V switch gear room creating a load path concern.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
PAL	96							The polar crane solenoid was installed (a modification) using duct tape and tie-wraps in a manner which resulted in overheating and failure of the solenoid coil, which resulted in a minor electrical fire.	
SON23	96							While lifting a section of the turbine hood, it became unbalanced and impacted a structural wall, damaging concrete.	
TP3	96							Multiple heavy load movements were performed over restricted areas.	
TP34	96							Heavy load exclusion areas were not properly documented or established.	
ZIO1	96							Fuel was moved without establishing adequate ventilation.	
ANO12	97							A nonconservative setpoint for load cell (read low) was noticed.	
ANO2	97							The crane was operated without establishing adequate ventilation.	
BRU1	97							A cask was lifted over safety related components without the cask valve covers installed.	
BRU12	97							Broken conductors were discovered on the main hoist cable of the refueling crane.	
BV1	97							Crane interlocks and stops were not tested prior to crane use.	
BV2	97							Spent fuel pool crane interlocks and stops were not tested prior to use.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BYR12	97				YES	YES		During the lift of a steam generator replacement runway section weighing approximately 26 metric tons (28.5 tons) located outside of containment, the runway section slipped about 4.6 meters (15 feet), came to an abrupt stop which caused the nylon rigging straps to fail. The runway section fell approximately 18 meters (60 feet) to the ground. Operator error was the most likely cause of the drop.	18 meters (60 feet)
CAL	97							A box weighing 204 kgs (450 pounds) was moved over the reactor vessel using the polar crane.	
CAT12	97							The wrong softener material was used for rigging protection during the movement of a spent fuel pool gate.	
CC1	97							Fuel was moved in the spent fuel pool without the charcoal adsorbers in service.	
CC1	97							Contrary to requirements, positive pressure was noticed in the spent fuel area during fuel movement in the spent fuel pool.	
CC2	97							A refueling machine was operated with its overload protection circuit bypassed.	
CLI	97							Multiple brake problems were experienced on the fuel building crane.	
DCC12	97							A new fuel shipping container was moved at a height that exceeded the maximum.	
FAR2	97							Loads were moved over the spent fuel pool without establishing proper ventilation.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
FITZ	97							Spent fuel was moved with the reactor building crane without adequate procedures.	
GG	97							A crane was operated having a "Do Not Operate" tag.	
HN	97							The load limit was exceeded when moving a load over the spent fuel pool.	
HN	97							Positive pressure was noted in the spent fuel building.	
HN	97							Ventilation for load movements was not adequate.	
IP2	97							Contrary to procedure, workers attempted to lift a cask pit cover and a recirculation pump at the same time.	
IP3	97							The spent fuel pool crane was operated without establishing adequate ventilation.	
IP3	97							Water intrusion into the brake coils resulted in their failure.	
MILL12	97							Heavy loads have been lifted over 480V vital switchgear multiple times.	
MILL2	97							A load weighing more than 817 kgs (1800 pounds) was moved over the spent fuel pool.	
MILL2	97							Overload protection was not provided for 23 cm (9 inches) of crane travel.	
MY	97							Loose electrical connections cause a problem with the trolley speed and intermittent operation of bridge speeds. This feature was not included in any surveillance testing.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
MY	97							During movement of the spent fuel pool bridge crane, it ran into a safety rail that had been moved by maintenance.	
MY	97							Crane tests were not performed prior to crane use.	
MY	97							During movement of the spent fuel pool bridge crane, its wheel guard cut a 480 V cable causing a short.	
MY	97							Mechanical stops were not placed correctly. The placement would have allowed a fuel bundle to strike the north wall of the pool.	
NMP12	97							A modification to add 10 rail splices to minimize cracking of hold down clip studs was added.	
PI1	97							Fuel was moved without adequate ventilation.	
PI1	97							The auxiliary building crane moved a cask while crane protective features were defeated.	
PI1	97							A spent fuel cask moved while two protective crane features were defeated by wiring errors.	
PI12	97							A load was moved over or close to safe shutdown equipment using a mobile crane.	
PI2	97							Reactor coolant pump motor was moved using an inadequate load path.	
PI2	97							A reactor coolant pump bracket and rotor weighing approximately 19 metric tons (21 tons) was moved over spent fuel without a safe load path, and without containment isolation.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
PIL	97							The load path was incorrect for the movement of a reactor head insulation package.	
PTB12	97							A crane hook collided with a ventilation duct.	
ROB	97							Multiple movements of casks were performed with the cask valve covers removed.	
SAL2	97							Inoperable crane locking devices were discovered.	
SUM	97							The service water pump removal load path included the corner of the service water building.	
SUS1	97					YES		While transporting a toolbox weighing approximately 1.8 metric tons (2 tons) using the reactor building crane auxiliary hoist, a sling parted, dropping the toolbox.	2.4 meters (8 feet)
SUS12	97							A 12.7 metric ton (14-ton) mobile crane tipped over when its boom was extended. Lack of training and supervisory oversight were listed as causes.	
TP34	97							Personnel were nearly hit by a manipulator crane when lowering fuel into the upender.	
TP34	97		YES					A personnel injury occurred that involved a crane.	
WAT	97							Refueling machine masts did not have overload protection.	
WAT	97					YES		A new fuel assembly was dropped during fuel movement in the spent fuel pool. The cause was not known.	Unknown
WC	97							The overload setpoint was not properly tested prior to use.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
ZIO2	97							The spent fuel pool crane was operated without establishing adequate ventilation.	
BV2	98							The spent fuel pool crane interlocks and stops were not tested prior to crane use.	
CAL	98							Heavy loads were moved over the RHR system.	
COL	98							The refueling bridge collided with a stationary hoist located next to the spent fuel pool.	
COL	98							Crane operators failed to verify temperature requirements prior to a load lift.	
DA	98		YES					A turbine building crane load shifted because of rigging issues. A worker was injured when the load struck him.	
DA	98					YES		A main generator exciter coupling weighing from 159 to 227 kgs (350 to 500 pounds) was dropped approximately 1.5 meters (5 feet) onto the turbine building floor due to improper rigging.	1.5 meters (5 feet)
DB	98							A polar crane lifted a ratchet plate weighing 0.9 metric ton (1 ton) over the refueling canal during fuel handling operations.	
DB	98							The power cable for the reactor service crane broke and became entangled on scaffolding.	
DB	98					YES		Rigging used to lift eddy current equipment came loose, resulting in a drop of the equipment 4.6 to 6.1 meters (15 to 20 feet).	4.6 to 6.1 meters (15 to 20 feet)
DB	98					YES		A portable transformer fell from a load that was being lifted.	1.5 meters (5 feet)

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
DB	98							A jib arm on the polar crane collided with a winch cable resulting in an equipment drop of approximately 61 meters (200 feet).	61 meters (200 feet)
DB	98							The polar crane control cable broke, resulting in the drop of the cable and pendant weighing a few hundred kgs (several hundred pounds).	43 meters (140 feet)
DC1	98							The polar crane was being used during power operations in jet impingements areas.	
DC1	98							The polar crane traveled over the jet impingement zone multiple times.	
DCC1	98							The load limit of the auxiliary hoist on polar crane was exceeded.	
FITZ	98							The reactor building crane was proposed to be used to move spent fuel.	
GG	98							A heavy load consisting of the core shroud inspection tool theta drive ring became partially disconnected from its strongback while being moved over irradiated fuel.	
GG	98				YES			A core shroud tool ring weighing 676 kgs (1490 pounds) became dislodged from its strongback during an accidental release of air through the reactor vessel.	Slight
HN	98							Ventilation requirements for load movements were not met.	
IP2	98							Contrary to procedure, two loads were lifted simultaneously in the chemical building.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
IP2	98							The polar crane was used to pull equipment through the airlock without an engineering evaluation.	
LIM12	98		YES					A worker was drawn into the turbine building hoist drum by his safety harness lanyard, resulting in serious injuries.	
MCG1	98							The licensee failed to perform a test of the reactor building crane auxiliary hoist after 100 hours of core load.	
MCG1	98							An auxiliary hoist load cell test was not performed prior to crane use.	
MIL1	98							New fuel assemblies were moved over the spent fuel pool using the reactor building crane. This movement was not previously analyzed.	
MIL3	98							A fuel handling crane limit switch surveillance test was not performed.	
MY	98							The fuel handling crane was not properly tested prior to use.	
PER	98							Tools and staging equipment were lifted over the reactor cavity.	
PER	98							A loaded crane in the ESW pump house was left unattended.	
PTB1	98							Interlocks and safety features were not fully tested prior to crane use.	
PV1	98					YES		During receipt of new fuel in the fuel building, a loaded container was dropped when the container lid separated from the bottom portion of the container.	5 cm (2 inches)

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
SEA	98							Ventilation for load movements was not adequate.	
SON23	98							There was a failure to monitor polar crane loads over irradiated fuel and safety related components.	
STP12	98					YES		A trailer used for snubber inspection was dropped from an elevation of about 31 cm (1 foot) on the Unit 2 Fuel Building truck bay. The leather glove that was used as a softener was insufficient to keep the sling from severing.	31 cm (1 foot)
TP3	98							Crane operators failed to follow procedures during fuel movement.	
WC	98							Polar crane snubbers were not properly inspected.	
BYR1	99							The manipulator crane locked up when removing fuel from the core because of gear failure.	
COL	99							The refueling floor crane hook measurements were not taken.	
CP1	99				YES			The Unit 1 reactor coolant pump motor weighing approximately 38 metric tons (42 tons) fell during a lift using an auxiliary hoist rigged to the polar crane. The slip occurred when the auxiliary hoist gearbox failed.	4.6 to 6.1 meters (15 to 20 feet)
CRY	99				YES			While using the polar crane, the lifting device for the reactor plenum was not attached properly and the load became cocked.	Negligible
DB	99							A heavy load was moved over the reactor vessel.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
DCC12	99							The weight of fuel assemblies increased without revising the accident analysis or load movement procedure.	
DRE23	99						YES	While lifting valves and pipe fittings from a laydown area, a mobile crane boom came to close to a 34 kV line. The line tripped and then reclosed. No injuries resulted.	
FAR2	99			YES				A failure of the polar crane primary height measuring system allowed a portion of the reactor lower internals to be exposed during the lift.	
MIL1	99				YES			A new fuel bundle being lifted by an auxiliary hoist on the reactor building crane continued to drift downward past its stop point until it came in contact with the refueling floor. No damage was done.	Not specified
NMP1	99							A reactor building hoist trolley connection failed due to fatigue.	
PAL	99							The maximum load weight for the reactor building crane was exceeded twice (greater than 125%) when lifting the reactor vessel head with additional lead shielding intact. A third lift was done at just under 125 percent of maximum weight.	
PI1	99							A procedure for a load lift did not exist.	
PI1	99							The reactor internals were lifted and transported using an inadequate load path.	
SEQ12	99					YES		A mobile crane tipped over while moving a cell cap to a compartment at the low level waste facility.	Short

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
SON1	99							The polar crane was not parked in an authorized area.	
SUM	99							A load cell surveillance test was not performed prior to crane operation.	
SUS1	99							The Unit 1 reactor building crane unexpectedly stopped during movement of a cask loaded with spent fuel. The event was caused by a defective hoist electrical drive controller.	
TRO	99							An ISFSI load movement lift height exceeded the maximum value in the procedure.	
TRO	99					YES		A mobile crane tipped over while lifting light poles because the outriggers were not extended.	Short
VY	99							The refueling bridge crane was operated with an inadequate procedure, resulting in a collision with the fuel mast causing minor damage.	
WC	99							During fuel movement, a fuel assembly was placed on top of another fuel assembly.	
BW12	00						YES	The loss of a 34 kV line to the river screen house, and subsequent tripping of the CW pumps was caused by a loaded mobile crane. There was no spotter for the crane operator. No damage was done to the crane, or the operator. The 34 kV line was damaged but not severed.	
BYR1	00							The spent fuel pool bridge crane, with a fuel assembly latched and moving horizontally, did not respond to its control system, resulting in a near collision. The main power breaker was opened to stop crane motion.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BYR12	00							A mobile crane lift of a reactor head cover and stand assembly became unbalanced. The load weight was much more than anticipated, 11 v.s. 8.2 metric tons (12.5 v.s. 9 tons). A load cell was not installed in the crane.	
BYR2	00							The spent fuel pool crane hoist lowered when the operator demand was for an upward movement. No damage occurred.	
DC1	00							The polar crane traveled over the jet impingement zone multiple times.	
DCC12	00							The auxiliary building crane was operated over the spent fuel pool with the load block energized during the load movement.	
DCC12	00							The auxiliary building crane was operated over the spent fuel pool without establishing adequate ventilation.	
DCC12	00							A hydraulic line on a mobile crane failed, resulting in an oil spill of many liters (several gallons).	
DCC2	00							Fuel was being moved in the spent fuel pool without establishing adequate ventilation.	
HN	00					YES		A radwaste canister filter dropped onto spent fuel racks caused by rigging deficiencies. No damage occurred.	A few meters (several feet)
MILL2	00							The spent fuel pool crane was operated without first performing surveillance tests.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
MY	00							While transporting the steam generator through the containment building, the steam generator collided with the top steel beam on a hydraulic gantry crane located outside the containment equipment hatch, and was knocked down, causing minor damage to the steam generator.	
OC	00		YES			YES		Two new fuel assemblies fell from their metal container onto the refueling floor of the reactor building because of rigging issues. One worker received a glancing blow when the fuel fell.	Not specified
PI2	00							There were repeated trips of the hoist overload relay that were caused by a poor procedure.	
PTB12	00							The primary auxiliary building crane hook collided with the multi-assembly transfer cask lifting yoke, causing minor damage.	
QC2	00							Fuel movement was performed in Unit 2 without a sufficient number of operable intermediate range radiation monitors.	
RS	00							The fuel storage building walls and crane rails were shifting, losing their proper alignment.	
SEA	00							The spent fuel pool bridge and hoist interlock was not tested for loads in excess of 0.96 metric tons(1.05 tons).	
BYR12	01							The fuel handling building crane operated intermittently, possibly caused by problems with the festoon cables.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BYR12	01							Eight welds on the bridge girder were found to be undersized, three of which were cracked. All eight welds were repaired.	
BYR12	01							A mobile crane tipped when lifting a load that was much heavier than predicted. In addition, the load cell was not functional. The load came in contact with the outside of the circulation water pump house. Only minor damage was done.	
BYR2	01		YES					While lifting a reactor stud bolt rack out of the reactor cavity, the stud rack caught on the handrail, pulling it loose. The handrail fell approximately 9.5 meters (31 feet) before striking a worker below. The worker was injured but equipment was not damaged.	
CC1	01							A crane hook rated at 2.7 metric tons (3 tons) was used to raise and transfer, in air, a container of waste having 91 curies of mixed radio nuclides weighing 3.2 metric tons (3.5 tons).	
CRY	01				YES			The spent fuel pool fuel handling hoist lowered on its own, caused by malfunctions of the main hoist switch.	Small
DCC2	01							The spent fuel pool crane load lift height was violated (crane height interlock was bypassed) when moving fuel over the spent fuel pool.	
DRE23	01							The crane condition was not maintained in accordance with design requirements (missing rail safety lugs, and the slow speed inching motor was not installed).	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
DRE23	01							A reactor vessel head assembly and carousel weighing 121 metric tons (132.5 tons) was lifted, while the crane was rated at 114 metric tons (125 tons). The crane load cell was reading low for years, indicating that the load was less than than actual.	
DRE23	01							The reactor building crane was docked over the Unit 2 spent fuel pool while performing work on the crane.	
MILL2	01							Heavy loads were moved over a safety-related pipe gallery many times using the cask crane which was not single failure proof.	
NA1	01					YES		The top nozzle and fuel assembly separated, leading to the drop of an assembly weighing less than 0.9 metric ton (1 ton). The break was caused by IGSCC.	3.7 meters (12 feet)
SON3	01					YES		A mobile crane weighing approximately 34 metric tons (37.5 tons) was dropped approximately 12 meters (40 feet) to the Unit 3 turbine bay floor when Kevlar slings failed. The mobile crane was severely damaged. The sling failure was caused by using inadequate rigging softener material.	12 meters (40 feet)
TP4	01					YES		A Link-Belt mobile crane weighing approximately 34 metric tons (37.5 tons) dropped approximately 20 cm (8 inches) to the Number 4 turbine building laydown area following the failure of a Kevlar sling. The sling separated because it was not properly protected at sharp corners. The mobile crane was inspected and found not to be damaged.	20 cm (8 inches)

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
BRP	02							The reactor building crane tripped while lifting a spent fuel cask out of the spent fuel pool. The emergency brake engaged, probably because of the hoist shifting from low speed to high speed.	
BYR1	02							During movement of the Unit 1 reactor head stud rack weighing approximately 4.1 metric tons (4.5 tons), it contacted the maintenance hoist, damaging the 480 V hoist bus bar. The Kevlar sling was also discovered to be over loaded and had stretched.	
BYR1	02							A liquid nitrogen bottle was damaged when it was hit by the auxiliary hook on the polar crane.	
BYR12	02							The spent fuel pool bridge crane hoist control pendant control went dead when attempting to move the crane. A programming malfunction occurred.	
BYR12	02							Turbine building crane #2 ran into Turbine building crane #1. Neither crane was loaded. The crane operator was not watching the signalman.	
BYR12	02							The fuel handling building crane was operated over the spent fuel pool with the hook energized.	
DC1	02							A low pressure turbine hood weighing 64 metric tons (70 tons) was lifted over the Unit 1 EDG room, and over a cable run for ASW pump 1-2.	
DCC12	02							The load interlock for the auxiliary building crane was not tested. The interlock prevents crane travel over the spent fuel pool. The interlock had not been tested since the new crane was installed.	

Table A1: Reported crane issues at U.S. nuclear power plants (continued)

PLANT NAME	EVENT DATE	DEATH	INJURY	RAD	LOAD SLIP	LOAD DROP	LOSS OF POWER	EVENT ABSTRACT	DROP/SLIP DISTANCE
PEB2	02					YES		The Unit 2 B recirculation pump motor weighing approximately 22 metric tons (24 tons) dropped to its motor stand when the hoist chain broke. Several causes were noted. The hoist was not tested to 125 percent of load prior to use. The hoist chain was not the correct material. In addition, plant conditions were not established prior to the load movement (a subsystem of the RHR shutdown cooling was not operable in case of a load drop).	25 cm (10 inches)
PTB12	02							An auxiliary building crane pendant cable became entangled with the spent fuel pool bridge during preparations for cask loading. The spent fuel pool crane shook, and operators observed sparks on the auxiliary crane. Power was lost to the auxiliary crane.	
SUR1	02							A manipulator crane lost power due to power cabling becoming caught on scaffolding, pulling the cabling from the crane.	
YR	02							A yard crane malfunctioned because of incorrectly set acceleration time limit switch for the main hoist, causing a trip of the control power circuit.	

Table A2: Plant name abbreviations

ANO	Arkansas Nuclear One	PEB	Peach Bottom
BELL	Bellefonte	PER	Perry
BF	Brown's Ferry	PI	Prairie Island
BRP	Big Rock Point	PIL	Pilgrim
BRU	Brunswick	PTB	Point Beach
BV	Beaver Valley	PV	Palo Verde
BW	Braidwood	QC	Quad Cities
BYR	Byron	RB	River Bend
CAL	Callaway	ROB	H. B. Robinson
CAT	Catawba	RS	Rancho Seco
CC	Calvert Cliffs	SAL	Salem
CLI	Clinton	SEA	Seabrook
COL	Columbia	SEQ	Sequoyah
COO	Cooper	SH	Shearon Harris
CP	Comanche Peak	SHO	Shoreham
CRY	Crystal River	SON	San Onofre
DA	Duane Arnold	STL	St. Lucie
DB	Davis-Besse	STP	South Texas Project
DC	Diablo Canyon	SUM	V. C. Summer
DCC	D.C. Cook	SUR	Surry
DRE	Dresden	SUS	Susquehanna
FAR	Joseph M. Farley	TMI	Three Mile Island
FER	Fermi	TP	Turkey Point
FITZ	James A. FitzPatrick	TRO	Trojan
FSV	Fort St. Vrain	VOG	Vogtle
FTC	Fort Calhoun	VY	Vermont Yankee
GG	Grand Gulf	WB	Watts Bar
GIN	Ginna	WAT	Waterford
HART	Hartsville	WC	Wolf Creek
HAT	Edwin I. Hatch	WNP	Washington Nuclear Power
HC	Hope Creek	YR	Yankee Rowe
HN	Haddam Neck	ZIO	Zion
HUM	Humbolt Bay		
IP2	Indian Point 2		
IP3	Indian Point 3		
KEW	Kewanee		
LAS	La Salle County		
LIM	Limerick		
MCG	McGuire		
MH	Marble Hill		
MIL	Millstone		
MONT	Monticello		
MY	Maine Yankee		
NA	North Anna		
NMP	Nine Mile Point		
OC	Oyster Creek		
OCO	Oconee		
PAL	Palisades		

Appendix B

EEG-74: Probability of Failure of the TRUDOCK Crane System at the Waste Isolation Pilot Plant (WIPP)

May 2000

INTRODUCTION

This appendix includes a report by the Environmental Evaluation Group (EEG) of New Mexico that discusses the probability of the failure of the TRUDOCK crane system at the Waste Isolation Pilot Plant (WIPP) located in New Mexico. The study used the failure data in NUREG-0612 and other sources to estimate the mean failure rate. Based on their calculations, a mean failure rate of 9.7 E-03 (1/year) was obtained. Calculations of confidence levels showed that there was a 71 percent likelihood that not more than one dropped load would occur in 103 years, and that there was a 95 percent likelihood that not more than one dropped load would occur in approximately 34 years. The EEG report predicted a much lower human error rate (e.g., a 25 percent contribution) than is experience in U.S. Navy reports or in the commercial U.S. nuclear power plant industry. This lower human error rate for the WIPP is attributed to greater training. In contrast, for Navy crane operation, human error rates between 90 and 95 percent were reported for 1996, 1997, and 1998.

EEG-74

PROBABILITY OF FAILURE OF THE TRUDOCK CRANE SYSTEM AT THE WASTE ISOLATION PILOT PLANT (WIPP)

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May 2000

EEG-74
DOE/AL/58309

PROBABILITY OF FAILURE OF THE TRUDOCK CRANE SYSTEM AT THE WASTE ISOLATION PILOT PLANT (WIPP)

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FOREWORD

The purpose of the New Mexico Environmental Evaluation Group (EEG) is to conduct an independent technical evaluation of the Waste Isolation Pilot Plant (WIPP) Project to ensure the protection of the public health and safety and the environment. The WIPP Project, located in southeastern New Mexico, became operational in March 1999 for the disposal of transuranic (TRU) radioactive wastes generated by the national defense programs. The EEG was established in 1978 with funds provided by the U. S. Department of Energy (DOE) to the State of New Mexico. Public Law 100-456, the National Defense Authorization Act, Fiscal Year 1989, Section 1433, assigned EEG to the New Mexico Institute of Mining and Technology and continued the original contract DE-AC04-79AL10752 through DOE contract DE-ACO4-89AL58309. The National Defense Authorization Act for Fiscal Year 1994, Public Law 103-160, and the National Defense Authorization Act for Fiscal Year 2000, Public Law 106-65, continued the authorization.

EEG performs independent technical analyses of the suitability of the proposed site; the design of the repository, its planned operation, and its long-term integrity; suitability and safety of the transportation systems; suitability of the Waste Acceptance Criteria and the compliance of the generator sites with them; and related subjects. These analyses include assessments of reports issued by the DOE and its contractors, other federal agencies and organizations, as they relate to the potential health, safety and environmental impacts from WIPP. Another important function of EEG is the independent environmental monitoring of background radioactivity in air, water, and soil, both on-site and off-site.

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ACKNOWLEDGMENTS

The authors wish to thank Jill Shortencarier for expert preparation of the manuscript. We thank our colleague Dale Rucker for helpful reviews and suggestions, and for contributing basic information. We also thank our colleague Robert H. Neill for helpful suggestions. The authors are also grateful to their colleagues at EEG (James K. Channell, Lokesh Chaturvedi, Ben Walker, William Bartlett) for reviews and suggestions. Special thanks are due to Matthew Silva for a valuable addition. We also thank Linda Kennedy for her expertise in improving the report and its references.

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SUMMARY

This probabilistic analysis of WIPP TRUDOCK crane failure is based on two sources of failure data. The source for operator errors is the report by Swain and Guttman, NUREG/CR-1278-F, August 1983. The source for crane cable hook breaks was initially made by WIPP/WID-96-2196, Rev. 0 by using relatively old (1970s) U.S. Navy data (NUREG-0612). However, a helpful analysis by R.K. Deremer of PLG guided the authors to values that were more realistic and more conservative, with the recommendation that the crane cable/hook failure rate should be 2.5×10^{-6} per demand. This value was adopted and used.

Based on these choices a mean failure rate of 9.70×10^{-3} (1/yr) was calculated. However, a mean rate by itself does not reveal the level of confidence to be associated with this number. Guidance to making confidence calculations came from the report by Swain and Guttman, who stated that failure data could be described by lognormal distributions. This is in agreement with the widely used reports (by DOE and others) NPRD-95 and NPRD-91, on failure data.

The calculations of confidence levels showed that the mean failure rate of 9.70×10^{-3} (1/yr) corresponded to a percentile value of approximately 71; i.e. there is a 71% likelihood that the failure rate is less than 9.70×10^{-3} (1/yr). One also calculated that there is a 95% likelihood that the failure rate is less than 29.6×10^{-3} (1/yr). Or, as stated previously, there is a 71% likelihood that not more than one dropped load will occur in 103 years. Also, there is a 95% likelihood that not more than one dropped load will occur in approximately 34 years.

It is the responsibility of DOE to select the confidence level at which it desires to operate.

PROBABILITY OF FAILURE OF THE TRUDOCK CRANE SYSTEM AT THE WASTE ISOLATION PILOT PLANT (WIPP)

1. INTRODUCTION

In March 1999, the Department of Energy began emplacing transuranic waste into the Waste Isolation Pilot Plant (WIPP). The facility is located in southeast New Mexico in bedded salt at a depth of 650 meters. The repository is designed to contain 176,000 cubic meters (850,000 drum equivalents) of contact-handled transuranic (CH TRU) waste and 7,100 cubic meters (8,000 canisters) of remote handled transuranic (RH TRU) waste. The contact handled waste will be shipped from various defense generator and storage sites throughout the nation in an NRC certified container called a TRUPACT II or in a shorter version called a HALFPACK. In preparation for shipping, fourteen drums of waste, two standard waste boxes, or eight overpack drums are lowered into each TRUPACT-II. An inner lid and an outer lid secure the top of the shipping container.

Upon arrival at the WIPP, the drums or boxes need to be unloaded from the shipping container. This will be done in the Waste Handling Building where there are two TRUDOCK cranes. The two cranes are six-ton overhead bridge cranes, and are capable of operating alone or in parallel. To unload each shipping container, the outer lid needs to be lifted (3520 lbs.) and the inner lid needs to be lifted (895 lbs.). Each is set to the side. Figure 1 shows that two seven drum arrays can be lifted and handled as a single unit. The lift is over two meters and the payload can weigh as much as 7,265 lbs. Assuming at least three lifting operations for each TRUPACT there would be 182,000 lifting operations to unload 850,000 drum equivalents of CH TRU waste or about 5200 lifting operations per year (1500 crane transfers/year x 3 lifts/TRUPACT-II) for the 35 year operational life of the facility.

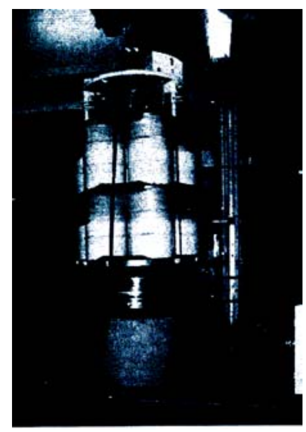

Figure 1. Lifting drums.

The DOE report, WIPP-WID-96-2196, Rev. 0, published in October 1996, studies and evaluates the possible frequency of failure of the TRUDOCK crane system, resulting in a dropped load and the loss of the drums' containment. The report turned to NUREG-0612 (July 1980) for failure data based on experience with U.S. Navy cranes in the 1970s.

However, the authors of WIPP-WID-96-2196, Rev. 0, 10/25/96, apparently had some concerns about using the data directly from NUREG-0612. The authors evidently turned for help to an independent source, Mr. R. Kenneth Deremer of PLG, an engineering consulting firm. Mr. Deremer's report is contained as Appendix A5 in WID-96-2196. According to Deremer's report, a preliminary version of the DOE report listed a failure "rate" of "2.0E(-5) per demand for crane cable/hook failures and cites NUREG-0612 as the basis for this value". Mr. Deremer is critical of that value, and then proceeded to his evaluation of a more realistic and more conservative value and he states that, "the crane cable/hook failure rate should be less than approximately 2.5E(-6) per demand"; that is a reduction of almost a factor of 10. Mr. Deremer makes the point that the NUREG-0612 data were compiled in the 1970s; and he states that the operating environment at WIPP is much less demanding than those for Navy cranes. He also mentions the aggressive inspection and maintenance programs at WIPP, "to assure the continuing reliability of the cranes." He believes that the failure rates could even be lowered, but states that "it is difficult to quantify this additional improvement".

In his summary Mr. Deremer strongly restates his recommendation of a choice in the data base for the crane cable/hook contribution "of the order of 2.5E(-6) per demand". Mr. Deremer's recommendation was adopted by the authors of WIPP-WID-96-2196. In the key table of that report, on the Crane System Cutset Descriptions, page A2-5, the Event Probability for the Crane Cable/Hook Breaks is listed as 2.5E(-6).

Support for the critical view by Mr. Deremer of the operating experience of Navy cranes may be seen by noting the relative frequencies of equipment failures vs. operator failures reported in "Navy Crane Incidents" (reports obtained from the U.S. Navy), for the recent years 1996, 1997 and 1998 (Table 1). The number of incidents associated with operator failure is an astonishing 90 to 95%.

TABLE 1

Frequencies of Navy Crane Incidents

Year	1998	1997	1996
Total no. of incidents	196	167	154
No. due to equipment failure/percentage	185/94.4%	16/9.6%	7/4.6%
No. due to operator failure/percentage	185/94.4%	151/90.4%	147/95.4%

Another source of information of hoisting and rigging incidents comes from a recent report by the Office of Oversight, U.S. Department of Energy, Washington, DC, 20585. The report is "Independent Oversight Special Study of Hoisting and Rigging Incidents within the Department of Energy", October 1996. The report covers a 30 month interval, from October 1, 1993 to March 31, 1996. The report states that "Human error is the major cause of hoisting and rigging incidents" (page 8).

This is similar to the data in the "Navy Crane Incidents" reports, with major causes of incidents due to operator rather than equipment failures.

In sharp contrast, in WIPP-WID-96-2196, for WIPP crane system experience, the operators are not the major cause of incidents. As the report states, "Crane operators and load spotters are required to be trained in safe crane operation; therefore it is felt that the WIPP crane performance will exceed the data presented in NUREG-0612, and the estimated failure frequency is felt to be conservative."

2. CALCULATIONS

2.1 Operator Errors

Operator errors are described in the table on page A4-6, of WIPP-WID-96-2196, Rev. 0. For convenience this table is reproduced (as Table 2) in this report, with some additions.

TABLE 2

Symbol	HEP*	Explanation of Error	Source of HEP	Page**
A1	$3.7x10^{-3}$	Improperly mate a connector, including failure to test the locking feature for engagement	Table 20-12.* Item (13), mean value.	20-28
B1	0.75	The operator repeating the action is modeled to have a high dependency for making the same error again. It is not complete dependence, because the operator moves to the second lifting leg and must physically push the locking balls to insert the pins.	Table 20-21. Item (4)(a), high dependence for different pins. Two opportunities (the second and third pins) repeat error is modeled as 0.5+(1-0.5)*0.5=0.75.	20-37
C1	$1.2x10^{-3}$	Checker fails to verify the proper insertion of the connector pins, and that status affects safety when performing tasks.	Table 20-22. Item (9), mean value.	20-38
D1	0.15	Checker fails to verify the proper insertion of the connector pins at a later step, given the initial failure to recognize the error. Sufficient separation in time and additional cues to warrant moderate rather than total or high dependence.	Table 20-21. Item (3)(a), moderate dependence for second check.	20-37
F1	$4.99x10^{-7}$	Failure rate if first pin improperly connected.	Product of above HEPs.	
a1	0.996	Given the first pin was properly connected.	1-F	
A2	$3.7x10^{-3}$	Improperly mate a connector, including failure to test the locking feature for engagement.	Table 20-12. Item (13), mean value.	20-28
B2	0.5	The operator repeating the action is modeled to have a high dependency for making the same error again. It is not complete dependence, because the operator moves to the second lifting leg and must physically push the locking balls to insert pins.	Table 20-21. Item (4)(a), high dependence for different pins. Only one opportunity for error (the third pin).	20-37
C2	$1.2x10^{-3}$	Checker fails to verify the proper insertion of the connector pins, and that status affects safety when performing tasks.	Table 20-22. Item (9), mean value.	20-38
D2	0.15	Checker fails to verify the proper insertion of the connector pins at a later step, given the initial failure to recognize the error. Sufficient separation in time and additional cues to warrant moderate rather than total or high dependence.	Table 20-21. Item (3)(a), moderate dependence for second check.	20-37
F2	$3.32x10^{-7}$	Failure rate if first pin improperly connected.	Product of above HEPs.	
FT	$8.31x10^{-7}$	Total failure rate due to human error.	$F_1 + F_2$	

* HEP stands for Human Error Probability.
** The source of the data is in a report by Swain and Guttman, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications”, August 1983, NUREG/CR-1278-F.

2.2 Crane System Cutset Descriptions

The table in page A2-5 of the WIPP-WID-96-2196 report lists all the basic events that can contribute to crane failures, including crane cable hook breaks, disk brake actuator failures, crane motor failures, etc., in addition to the operator errors. The calculations of the "cutset probabilities" (chances of failing) are given in detail. Of the thirteen listed cutset probabilities, only the first four need to be considered, since the remaining nine are orders of magnitude smaller.

Table 3 is an abbreviation of the table in page A(2-5), and it lists the four contributing components to the cutset probabilities. Note that the failure rate for a crane cable hook break is listed as 2.50 x 10^{-6} (1/demand), the value recommended by Mr. Deremer. The mean failure rate, per demand, due to operator error, is 8.31 x 10^{-7} (see in Table 3). As indicated in Table 3, the reduced sum of the probability of failure is 9.70 x 10^{-3} (1/yr), or approximately, one failure every 103 years.

TABLE 3

Cutset Probabilities

	Cutset Number	Ref.	Page	Failure Mode	Failure Rate mean, per demand	Number Crane Transfers/ yr App A_2, A_{2-3}	Event Prob. (Cutset prob.)
Equipment Failure	1	Appendix A_5	3	Crane B cable hook breaks	$2.5x10^{-6}$	$1.456x10^{3}$	$3.64x10^{-3}$
	2	Appendix A_5	3	Crane A cable hook breaks	$2.5x10^{-6}$	$1.456x10^{3}$	$3.64x10^{-3}$
Operator Error	3	Table 2 (this report)		Improper connection due to Operator Error*	$8.31x10^{-7}$	$1.456x10^{3}$ (Crane B)	$1.21x10^{-3}$
	4	Table 2 (this report)		Improper connection due to Operator Error*	$8.31x10^{-7}$	(Crane A) $1.456x10^{3}$	$1.21x10^{-3}$

$$P = \sum_{1}^{4} E.P. \text{ = Reduced Sum of Probability of Failure}$$

= 9.7x10-3 (1/yr)
Or approximately one failure every 103 years.

*Note that the contribution of Operator Error is only about 25% of the total.

2.3 Use of Confidence Levels

The calculation in Table 3 of the probability of failure doesn't tell the whole story. One also wishes to know the confidence level that is associated with the failure rate of 9.70×10^{-3} (1/yr). It is helpful to follow the recommendations of the Nuclear Regulatory Commission to include mean estimates and to "take into account the potential uncertainties that exist so that an estimate can be made on the confidence level to be ascribed to the quantitative results." This quotation is taken from the Nuclear Regulatory Commission (NRC 1986). EEG makes the same recommendation, and a calculation of confidence levels is made in this report.

A suggestion of the distribution of HEPs (see Table 2) is made by Swain and Guttman (NUREG/CR-1278-F, page 7-1, page 2-18) to use lognormal distributions. Another helpful source on this matter are the reports NPRD-91 and the more recent NPRD-95, by Denson, Chandler, Crowell, Clark and Jaworski, 1994, "Nonelectronic Parts Reliability Data." Both reports NPRD-91 and NPRD-95 have been used as sources of failure data by DOE in their recent reports: WIPP/WID-96-2178, Rev. 0, July 1996 and WCAP-13800, February 1994 (Preliminary Draft Report).

Both NPRD-91 and NPRD-95 state that all listed failure rates "estimate" the expected failure rates, and that the "true" values lie in some confidence intervals about these estimates. The following statement is a quote from NPRD-91 (Denson et al. 1991), page 1-6:

> "To give NPRD-91 users a better understanding of the confidence they can place in the presented failure rates, an analysis was performed on the variation in observed failure rates. It was concluded that, for a given generic part type, the natural logarithm of the observed failure rate is normally distributed with a sigma (σ) = 1.5. This indicates that 68 percent of actual failure rates will be between 0.22 and 4.5 times the mean value. Similarly, 90% of actual failure rates will be between 0.08 and 11.9 times the presented value."

This is to state that if one wishes to include 90% of all the failure rates, one must include a range of values that somewhat exceeds two orders of magnitude [11.9/0.08 = 148] . Under these circumstances, representing the failure rate by a mean value alone disregards relevant information.

2.4 Lognormal Calculations

A general form for the lognormal distribution with the two parameters, μ, F is given by (Aitchison and Brown, 1969):

$$d\Lambda(x) = \frac{1}{(x\sigma\sqrt{2\pi})}\exp\{-\frac{1}{2\sigma^2}\left(\log x - \mu\right)^2\}dx$$

where Λ is the cumulative distribution function (CDF).

The median of the distribution is given by: $x_{md} = e^{\mu}$ (1)

The mean is given by: $x_{mn} = e^{\mu+(1/2)\sigma^2}$ (2)

According to the NPRDs (Denson et al., 1991, 1994) σ is taken as equal to 1.5.

from (2): $\mu = \ln[e^{-(1/2)\sigma^2} \bullet x_{mn}]$

since σ = 1.5; $e^{-(1/2)\sigma^2} = e^{-1.125} = 0.3247$

thus: $\mu = \ln[0.3247 \cdot x_{mn}]$ (3)

The values of (EP), the Cutset Event Probabilities, are listed in the right most column of Table 3.
Let $x = 10^3 \bullet (EP)$; the values of x are listed in Table 4.

TABLE 4
Calculations of the Values of μ

Cutset Number	(From Table 3) x_{mn} = 103 •(EP) = $e^{\mu+(1/2)\sigma^2}$	(From Equation 3) e^{μ} = (0.3247x_{mn})	μ
1	3.64	1.1819	0.1671
2	3.64	1.1819	0.1671
3	1.21	0.3929	-0.0342
4	1.21	0.3929	-0.9342

Table 4 can be summarized as follows:

Table 5
Values of the Parameters μ, σ

P_i	μ_i	σ_i
1	0.1671	1.5
2	0.1671	1.5
3	-0.9342	1.5
4	-0.9342	1.5

The failure distribution, P, can be expressed as follows:

$$P = 10^{-3} \sum_{i=1}^{i=4} P_i \qquad (4)$$

$$dPi(x) = \frac{1}{(x\sigma_i\sqrt{2\pi})} \exp[-\frac{1}{(2\sigma_i^2)}(\log x - \mu_i)^2]dx \qquad (5)$$

The failure distribution, P, has been expressed as the sum of four lognormal random variables, P_i. The factor 10^{-3} is introduced to cancel the 10^3 used in the columns of Table 4. The methods used to compute the failure distribution functions are described in detail in the Appendix.

Table 6 lists the percentile values for the approximating probability distribution of the grand total of the four random variables.

TABLE 6
Percentiles and Probability Values of the Grand Total of the Four Random Variables

Percentile	Probability x 10^3 (1/yr)
0.5	0.65
1.	0.8
5.	1.4
10.	1.85
20.	2.8
50.	5.8
Mean = 71	9.70
80	12.8
90	20.
95	29.6
99	65.5
99.5	89.5

TABLE 7
Comparison of Means and Variance

True Mean	9.701
Approximating Mean	9.698
True Variance	249.83
Approximating Variance	246.51

For the grand total of the four random variables the approximating and true means and variance are listed in Table 7 without the factor (10). The values of the approximations are close to the $_{-3}$ true values. This indicates that the approximations for the probability values listed in Table 6 have relatively small errors.

3. DISCUSSION

The data in Table 6 for the probability and the percentiles have been plotted on "probability-log" graph paper; see Figure 2. Some statements may be made, based on Figure 2 or Table 6.

(a) The mean failure rate is 9.70×10^{-3} (1/yr), and corresponds to a percentile value of approximately 71, i.e. there is a 71% likelihood that the failure rate is less than 9.70×10^{-3} (1/yr).

(b) At the 95 percentile, the probability is slightly less than 30×10^{-3} (1/yr) (actually 29.6 from Table 6); i.e. there is a 95% likelihood that the failure rate is less than 29.6×10^{-3} (1/yr).

(c) The above statements may be recast in another way:

There is a 71% likelihood that not more than one dropped load will occur in 103 years. Also, there is a 95% likelihood that not more than one dropped load will occur in approximately 34 years. One may calculate the corresponding time intervals for lower and higher levels of likelihood. Which level of likelihood does one select? That choice is the responsibility of DOE to make.

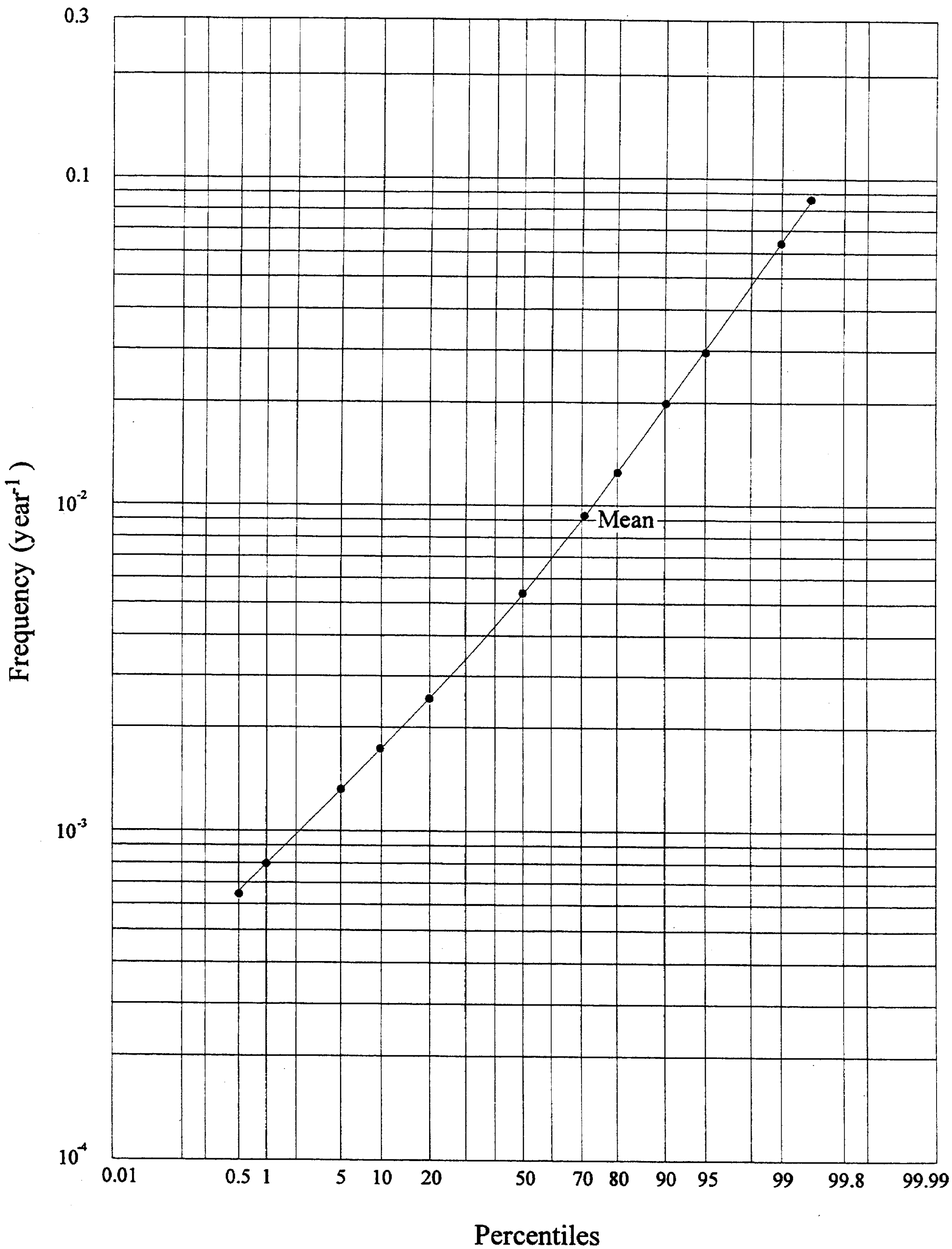

Figure 2. Cumulative distribution function for probability of failure of the TRUDOCK crane system

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LIST OF ACRONYMS

EEG	Environmental Evaluation Group
EP	Event Probabilities
NPRD	Nonelectronic Parts Reliability Data
NRC	Nuclear Regulatory Commission
DOE	United States Department of Energy
WID	Waste Isolation Division
WIPP	Waste Isolation Pilot Plant

APPENDIX

APPENDIX

This appendix describes how we numerically approximated the density of the random variable $x = \sum_{i=1}^{4} x_i$ where each x is distributed independently of x_j, $j \neq i$ and is log normal with parameters μ_i, σ_i. The distribution of a *sum* of two independent random variables is the *i i* convolution of the two distributions. But convolution in the time domain corresponds to multiplication in the frequency domain. This allows us to compute the distribution that we want by taking the Fourier transform of each density, multiplying them, and then inverse Fourier transforming.

Transform Methods

We want to compute the density function for a random variable that is the *sum* of two independently distributed random variables with known densities.[1] We use the following theorem:

Theorem: Let *x* be a continuously distributed random variable with density *f(x)*, and let *y* be a continuously distributed random variable with density *g(y)*. Let *x* and *y* be independently distributed. Then the random variable $z = x + y$ is distributed with density h(z) given by the *convolution* of *f* and g, which is defined by

$$h(z) = \int f(u)g(z-u)du$$

A related theorem governs discrete approximations to continuously distributed random variables.

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[1]The mathematical theorems can be found in many books on operational mathematics. For example, see R.A. Gabel and R.A. Roberts, *Signals and Linear Systems*, Wiley, 1973.

Theorem: Let x be a random variable that takes values on the set $X = [x_0, x_1, \ldots, x_{T-1}]$, with density f_t = Prob[$x = x_t$]. Let y be another random variable that takes values on the *same* set X, with density g_t = Prob[$y = x_t$]. Let z be the random variable $z = x$ + y, and let x and y be distributed independently. Then z has density h with

$$h_t = \Sigma f_k g_{t-k}$$

where h = Prob[$z = z_t$], and where z resides in the discrete set $Z = [2x_0, \ldots, 2x_{T-1}]$.

The next useful result is that the Fourier transform of a convolution is the *product* of the Fourier transforms of the two sequences being convoluted. The *Fourier transform* of a sequence $\{x_t\}_{t=0}^{t=T-1}$ is defined as the sequence of complex numbers given by

(1) $$x(\omega_j) = \Sigma_{t=0}^{T-1} x_t e^{-i\omega_j t}$$

where ω = $2\pi j/T$ and j = 0,1,...,T -1. The *inverse Fourier transform* is given by $_j$

(2) $$x_t = T^{-1}\Sigma_{j=0}^{T-1} x(\omega_j) e^{i\omega_j t}$$

Equations (1) and (2) constitute the basic Fourier transform pair. Notice that the inverse Fourier transform of the Fourier transform is the original sequence.

The key theorem for us is:

Theorem: The Fourier transform of the convolution of two sequences $\{x_t\}$ and $\{y_t\}$ is the
product of their Fourier transforms x(ωj) y(ωj).

We apply this theorem as follows. For each of two continuous distributions, *(f,* g), the probability laws for *(x ,*y), respectively, we put down a discrete 'grid' of points $X = [x_0 ,...,x_{T-1}]$ on the real line, with the points spaced close enough together and over a sufficiently large set to approximate each continuous distribution well. Then we used *(f,* g) to generate approximating discrete probability distributions for *(x,* y). For computational consistency, we used the *same* grid for each random variable under study. We chose the grid carefully to make sure that each random variable as well as the relevant sums were well approximated by the procedure. For each approximating distribution $\widetilde{f}_t$ and $\widetilde{g}_t$, we computed the Fourier transform *f (*ωj) and g(ωj). Then we computed the Fourier transform of $\{\widetilde{h}_t\}$, the approximating distribution of the sum *x* + y, as

$$h(\omega) = f(\omega_j)g(\omega_j).$$

To compute the approximate density of *x + y,* $\widetilde{h}_t$*,* we then inverse Fourier transformed *h* (ωj):

$$\widetilde{h}_t = T^{-1}\Sigma_{t=0}^{T-1} h(\omega)e^{i\omega_j t}$$

Computational Details

We implemented these calculations using the *Fast Fourier Transform* (FFT) and the associated inverse transform, the IFFT. We used the computer language MATLAB on a Dell 450 MHz PC with 128 x 3 K of memory. This permitted us to put down very large and fine grids. We used one (inconsequential) approximation: each time a convolution is computed, the FFT in effect *truncates* the grid on which the relevant sum is distributed, and restricts it to the same domain on which the original two distributions are defined. In particular, the density of the sum is computed only on the *same* domain $X = [x_0 ,x_1 ,...,x_{T-1}]$, rather than on the true domain $Z = [2x_0 ,...,2x_{T-1}]$. To control the error resulting from this approximation, we select the grid set X very carefully to make sure that it covers the region where the pertinent *x, y,* and sum *z* = *x* + *y* have appreciable positive probability.

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Appendix C

Independent Oversight Special Study of Hoisting and Rigging Incidents Within the Department of Energy

October 1996

INTRODUCTION

This appendix includes a U.S. Department of Energy (DOE) report, “Independent Oversight Special Study of Hoisting and Rigging Incidents Within the Department of Energy,” which was issued in 1996. The DOE report presents the results of an analysis of hoisting and rigging (H&R) incidents, covering the period from 1993 to 1996. DOE defined H&R to include the raising, moving, and unloading of materials, either by large power-lifting equipment, such as cranes and forklifts, or by smaller, light duty manual and power-operated equipment, such as hoists, chainfalls, and block and tackle. Human error, whether directly associated with supervisors or equipment operators represented approximately 94 percent of H&R incidents. Factors not related to human performance, such as equipment failure and weather, were responsible for only 6 percent of H&R incidents. Inattention to detail (56 percent) and not following procedures (28 percent) account for 84 percent of H&R incidents caused by personnel error.

INDEPENDENT OVERSIGHT SPECIAL STUDY OF HOISTING AND RIGGING INCIDENTS WITHIN THE DEPARTMENT OF ENERGY

October 1996

Office of Oversight
Office of Environment, Safety and Health
U.S. Department of Energy
Washington, DC 20585

ACKNOWLEDGEMENT

The Office of Oversight gratefully acknowledges the assistance from members of the Department of Energy Hoisting and Rigging and Construction Safety Advisory Committees for their review of and comments on the materials presented in this report. Their timely contribution ensured technical accuracy in the interpretation of the study results.

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EXECUTIVE SUMMARY

This report presents the results of an analysis of Department of Energy (DOE) hoisting and rigging (H&R) incidents, covering the period beginning October 1, 1993, and ending March 31, 1996. The study, initiated at the request of the Assistant Secretary for Environment, Safety and Health, was performed in response to concerns over the safety of H&R operations, and the perception that accidents were occurring with greater frequency. The purpose of this study was to determine whether additional oversight of H&R operations is warranted. The results of this effort will be combined with information from other independent oversight initiatives, to determine the effectiveness of the Department's overall safety management program, and to develop strategies to combat systemic problems that hinder the attainment of satisfactory safety performance.

Hoisting and rigging includes the raising, moving, and unloading of materials, either by large power-lifting equipment, such as cranes and forklifts, or by smaller, light duty manual and power-operated equipment, such as hoists, chainfalls, and block and tackle. These activities, which pervade work performed throughout the DOE, have long been viewed as an area presenting significant safety challenges.

An H&R incident is defined as an unsafe situation that either 1) required immediate cessation of the activity, 2) resulted in an accident, or 3) almost incurred an accident (i.e., a near miss). In the past five years, H&R incidents have resulted in fatalities, personal injuries, and property damage — accidents. Since October 1993, three out of every four H&R incidents resulted in an accident where personal injury, property damage, or both were incurred. Despite management attention to H&R operations in the aftermath of these events, incidents continued without a pronounced trend. The activities and operations that constitute the DOE H&R process have not basically changed, and management has not been successful in improving the process.

Half of all H&R incidents are associated with the use of crane equipment, and almost a third of all H&R incidents involve forklifts. Seventy-four percent of crane incidents, and 90 percent of forklift incidents, resulted in accidents. Inattention to detail, closely followed by deficiencies in work organization and planning, is the leading cause for crane incidents. Inattention to detail and procedures not used or used incorrectly are responsible for most forklift incidents. Deficient work planning and organization, and inadequate or defective engineering design or configuration contribute to almost half of all incidents involving "other" H&R equipment (i.e., manual and power-operated hoists, chainfalls, and block and tackle).

The strong relationship identified in this review between the root causes of H&R incidents and the type of equipment used provides a tool that can be used to improve H&R safety performance. For example, as the Department transitions from production to environmental restoration, greater use of subcontractor-operated mobile cranes is anticipated. Greater oversight of subcontractor operations by line management that emphasizes the importance of attention to detail and effective work organization and planning will improve the safety of their operations. Implementation of effective strategies to address incidental use of heavy-duty H&R equipment, such as forklifts, will contribute to reducing a large proportion of H&R accidents.

INDEPENDENT OVERSIGHT SPECIAL STUDY OF HOISTING AND RIGGING INCIDENTS WITHIN THE DEPARTMENT OF ENERGY

1.0 INTRODUCTION

The Office of Oversight analyzed hoisting and rigging incidents that occurred within the Department of Energy between October 1993 and March 1996.

This report presents the results of an analysis of Department of Energy (DOE) hoisting and rigging (H&R) incidents during the 30-month period beginning October 1, 1993, and ending March 31, 1996. It is one of numerous independent assessment activities performed by the Office of Oversight. The information presented in this report will be combined with the results of other independent oversight efforts, including site-specific evaluations and special studies of important topical areas, to evaluate the effectiveness of the Department's overall safety management program and identify areas for further evaluation.

BACKGROUND

Hoisting and rigging activities present significant safety considerations.

Hoisting and rigging activities include raising, moving, and unloading materials, either by large power lifting equipment, such as cranes and forklifts, or by light duty manual and power-operated equipment, such as hoists, chainfalls, and block and tackle. These activities are viewed as presenting significant safety considerations. This view is shared not only by the Department, but by other Federal government organizations and private industry. Recent events, observations, and findings from various inspections by the DOE, as well as general perceptions, have heightened the awareness and concern for the safety of H&R operations within the Department. This study was performed in response to these concerns at the request of the Assistant Secretary for Environment, Safety and Health.

Many DOE activities involve technologies, equipment, and processes that are unique to a specific program or facility. However, H&R operations do not vary significantly among the various DOE sites. Depending on the lifting source, load-lifting cables may be used (such as in the case of crane operations) to raise, suspend, and move materials that are generally secured by ropes, chains, or synthetic web straps. H&R tasks pervade work performed throughout the DOE complex in the construction, operation and maintenance, decommissioning and decontamination, and environmental restoration phases of a facility or project. Consequently, a better understanding of safety performance of H&R operations can have wide application.

Hoisting and rigging incidents led to a "lessons learned" workshop in 1994 to improve safety performance.

Large machinery (e.g., cranes), suspended loads, and substantial hazards characterize H&R operations. The safety of H&R tasks is dependent on sufficient supervision, proper hazard analysis and work planning, and appropriate selection, operation, and maintenance of equipment. Within the past five years, the safety performance of H&R operations has been marred by events that resulted in serious injuries to workers, substantial property damage, and fatalities at the Idaho National Engineering Laboratory in May 1991 and at the Oak Ridge Reservation in November 1992. These events heightened the need for increased management attention to H&R operations.

Recurring incidents indicate that safety in hoisting and rigging requires further improvement.

In April 1994, the Department sponsored the Hoisting and Rigging Lessons Learned Workshop, attended by DOE managers, supervisors, and staff, and contractor personnel. The workshop was devoted entirely to examining the knowledge gained from recent hoisting, rigging, and materials handling incidents for purposes of improving the safety of future operations. However, H&R safety performance continues to be of concern as incidents and accidents recur.

Throughout this report, reference is made to H&R incidents as distinct from accidents. An incident is defined as an unsafe situation that either 1) required immediate cessation of the activity, 2) resulted in an accident, or 3) almost incurred an accident (i.e., a "near miss"). An accident is a situation that results in fatality, personal injury, or property damage. The term "accident" does not include "near misses."

PURPOSE AND OBJECTIVES

Causes and trends were examined to identify actions to improve safety performance.

This analysis was conducted to better understand H&R incidents throughout the DOE complex in order to determine whether additional oversight of H&R operations is warranted. If focused correctly, additional oversight of H&R operations may improve safety performance by uncovering information helpful in combating systemic problems that hinder the effectiveness of safety management throughout the Department. Accordingly, the study is intended to:

- Determine the principal causes of H&R incidents.
- Identify significant trends in H&R incidents and accident consequences.
- Identify potential actions to prevent or limit H&R incidents.

ORGANIZATION OF THE REPORT

The technical approach, including sources of information and analytical techniques used, is provided in Section 2. Section 3 presents the results of the analysis. Conclusions are contained in Section 4. Appendix A contains information on the analytical and statistical methods used. Appendix B lists those involved in developing the report.

2.0 APPROACH

This section describes the method used to examine the safety performance of H&R incidents throughout the Department. It presents information on the data analyzed and techniques employed to address the study objectives.

OVERVIEW OF APPROACH

Incidents reported in the Occurrence Reporting and Processing System were examined.

The DOE Occurrence Reporting and Processing System (ORPS) served as the principal information source for incidents relating to H&R operations. Various analytical techniques, including Pareto analysis, process control, regression analysis, and other statistical methods were applied to information on H&R incidents extracted from occurrence reports to analyze root causes and identify meaningful trends.

ANALYTICAL TECHNIQUE

A narrative search was performed on data contained in ORPS to extract an initial set of 491 occurrence reports, corresponding to the October 1, 1993, to March 31, 1996, period, describing incidents related to H&R[1]. An H&R incident was considered relevant for further analysis if it:

- Occurred during hoisting and rigging operations, or the use of hoisting and rigging equipment, as defined in the U.S. Department of Energy Hoisting and Rigging Handbook

AND if it:

- Resulted in unsafe or improper conditions that necessitated the immediate suspension of the hoisting and rigging operation for any period of time, led to a near miss, or caused an accident.

Occurrence reports documenting the identification of suspect or counterfeit parts in H&R equipment were excluded if the part in question did not contribute to an operational incident. Suspect or counterfeit parts were not reported as the root cause for any of the incidents analyzed. Similarly, incidents pertaining to skin, clothing, and equipment contamination during H&R operations were excluded unless they created a contamination incident.

Although information in the Occurrence Reporting and Processing System has some flaws, the hoisting and rigging incidents reported there warrant attention.

Inconsistencies and ambiguities were identified in the assignment of root causes to incidents, as reported into ORPS. For example, it was not clear from ORPS occurrence reports why the root causes of certain incidents were attributed to management or poor work environment while

[1] See Appendix A for a description of narrative search technique.

similar incidents were attributed to inattention to detail. This lack of clarity in root cause determination is consistent with deficiencies in occurrence reporting identified by the Office of Oversight in November 1995, when it was reported that some personnel responsible for occurrence reporting are not adequately trained in the analysis of root causes[2]. Because this study used only information readily available from ORPS, no interviews were conducted to resolve these issues. However, these areas are identified as warranting attention to improve the utility of ORPS[3]. Despite these shortcomings, the number of H&R incidents recorded in ORPS deserves attention, especially because H&R accidents can have severe consequences.

There were 131 relevant hoisting and rigging incidents between October 1993 and March 1996.

Keeping in mind these issues, along with the variability in terminology used to report H&R incidents, application of this technique and the associated criteria produced 131 relevant H&R occurrence reports for the 30-month period. Information contained in these reports on the characteristics of each H&R incident was used to construct the database analyzed. As discussed in Appendix A, this database represents approximately 41 percent of the total number of relevant H&R occurrences contained in ORPS; thus it provides a basis for extrapolating and making inferences to the entire population with less than a five percent error at the 95 percent level of confidence[4].

3.0 RESULTS

This section summarizes the study results, including types of incidents and root causes and trends.

OVERVIEW OF SAFETY PERFORMANCE

After a fatal accident in November 1992, Departmental hoisting and rigging activities were curtailed.

The November 1992 H&R fatality at the Oak Ridge Reservation K-25 Gaseous Diffusion Plant, along with other less serious incidents at this and other sites, precipitated a suspension of H&R operations at Oak Ridge beginning in April 1993, which lasted approximately three months. Similar curtailments in H&R activities were implemented elsewhere in the Department.

Despite various management actions, incidents continue to recur.

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[2]See report entitled Independent Oversight Special Study of Occurrence Reporting Programs within the Department of Energy, November 1995, p. A-4.

[3]Root causes analyzed were those assigned to incidents contained in occurrence reports. No adjustments were made to reconcile inconsistencies. Definitions for root causes are defined in DOE Order 5000.3B (1/19/93-10/29/95), and its successor, DOE Order 232.1 (10/30/95-9/25/99, both entitled Occurrence Reporting and Processing of Operations Information.

[4]See Appendix A for inferential strength of sample.

After this period, beginning in January 1994, the number of H&R incidents reported throughout the Department followed a generally downward trend, reaching a low point in June 1994, shortly after the DOE Hoisting and Rigging Lessons Learned Workshop held in April that same year that was designed to improve H&R safety performance. While the Oak Ridge fatality, the subsequent cutback in H&R operations Department-wide, and the DOE workshop may have had some effect on DOE H&R activities that contributed to improved safety awareness, their relationship to the reduced number of reported H&R incidents cannot be verified. In any event, this sitewide trend reversed itself shortly after the DOE workshop, and by August 1994 H&R incidents began increasing to a generally higher level, where it remains today without exhibiting a discernable upward or downward trend.

The trend since 1993 is depicted in Figure 1. While random variations in safety performance are expected, there may be other factors that influence the time interval between incidents, including work stoppages and additional caution for a period following an event; these are discussed in Appendix A. Despite fatalities, suspended operations, and the workshop, there has been no statistically significant change in the frequency of reported H&R occurrences or in the H&R process within the Department.

More incidents can be expected as site cleanup efforts accelerate.

The consequences of H&R incidents can be significant. Approximately three fourths of H&R incidents resulted in an accident where personal injury, property damage, or both was incurred. Although the available information was limited, it appears that in at least 4 percent of the accidents, property damage alone exceeded $25,000 per accident. While H&R operations and incidents are common to many activities, including testing, fuel movement, and weapons management, the number of H&R operations and incidents is likely to increase as site cleanup efforts accelerate.

DISTRIBUTION OF HOISTING AND RIGGING INCIDENTS

Most hoisting and rigging incidents result from operations involving cranes or forklifts.

Lifting operations utilizing crane equipment generally involve complex maneuvers with large suspended loads. Fifty percent of all H&R incidents analyzed, and 51 percent of all H&R accidents, involved cranes. Forklifts were associated with 31 percent of all H&R incidents and 38 percent of all accidents. Less than 20 percent of all incidents, and 11 percent of all accidents, are associated with "Other" types of H&R equipment, such as manual and power-operated hoists, chainfalls, and block and tackle. The distribution of incidents and accidents associated with H&R operations is summarized in Table 1.

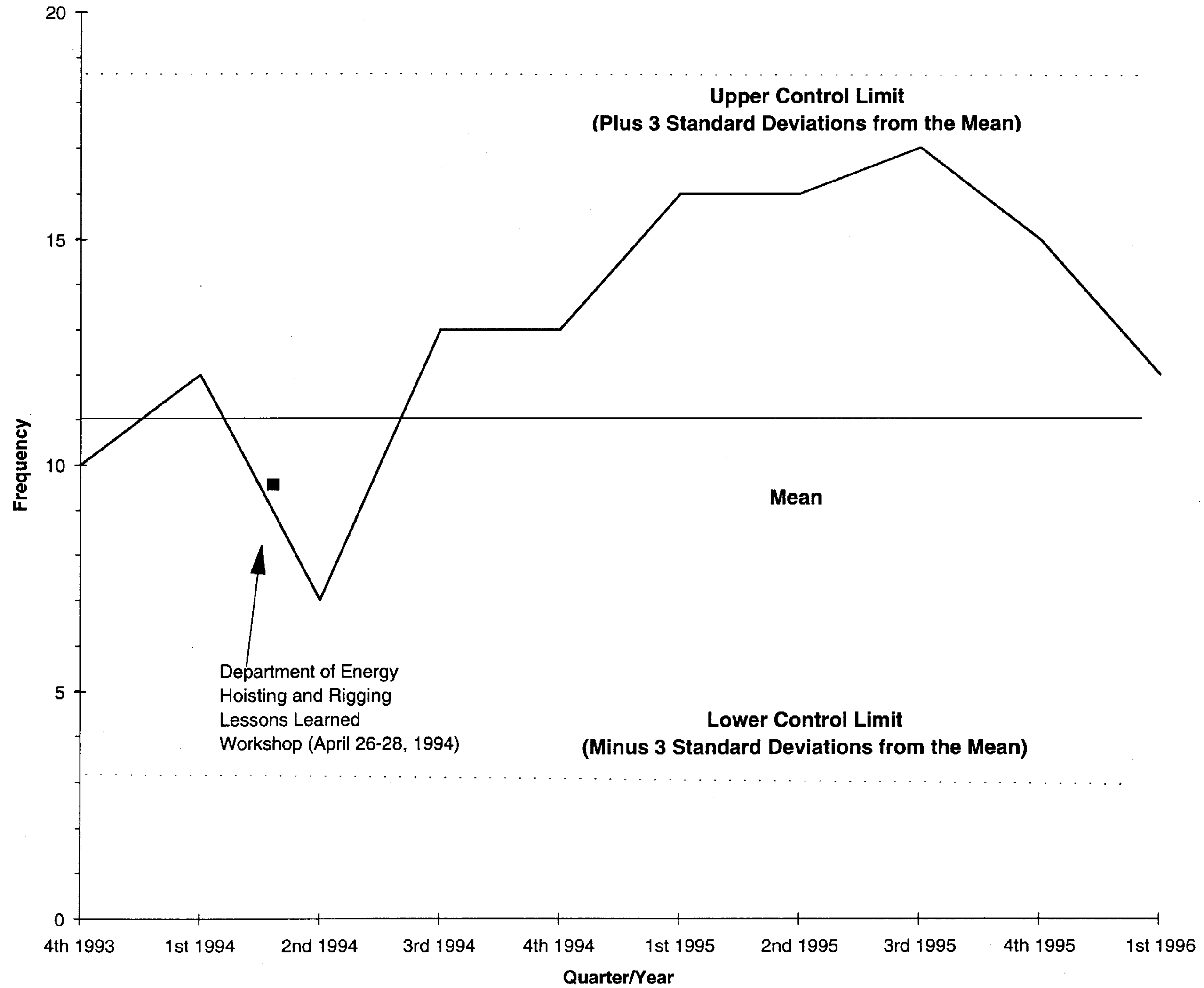
Incidents involving forklifts resulted in an accident more often than those involving cranes or "Other" hoisting equipment. As shown in Table 1, about one third of all incidents involved forklifts, and

Table 1. Distribution of Hoisting and Rigging Incidents and Accidents

Equipment	Number of Incidents	Number of Accidents	Incidents as a Percent of Total*	Accidents as a Percent of Total*	Accidents as a Percent of Incidents*
Crane	66	49	50%	51%	74%
Forklift	40	36	31%	38%	90%
Other**	25	11	19%	11%	44%
Total	131	96	100%	100%	73%

*Rounded to the nearest whole number.
**Includes manual and power-operated hoists, chainfalls, and block and tackle.

Figure 1. Frequency of Hoisting and Rigging Incidents*



Most forklift incidents result in an accident.

90 percent of all forklift incidents resulted in an accident. Generally, tasks using cranes, especially mobile units, involve several people in addition to the operator, such as spotters and signalers. Forklift operations usually require only the operator. Crane operators can generally see the load fairly well, and the additional personnel involved in crane operations augment attentiveness. Consequently, crane operators are better able to control a lift and curtail or suspend operations to avoid an accident. Also, cranes usually transport their loads at a height that is free of obstructions. Forklifts often encounter traffic, terrain, and other physical obstacles during load transportation; these appear to contribute to incidents.

Unlike cranes, forklifts are often used for incidental, non-repetitive tasks.

Discussions with H&R managers and supervisors from DOE, contractor, and subcontractor organizations verified that crane operation is generally a dedicated job, whereas forklifts are operated at various times by a variety of personnel in order to accomplish incidental tasks—that is, a forklift is a "tool." For example, forklifts can both tow like a tractor (which is not considered a H&R-related operation) and hoist like a crane. In this latter (and unconventional) application, a forklift is commonly referred to as a "free-rigger." The forklift tines are used to raise, suspend, and move materials secured by rigging (e.g., ropes, chains, or synthetic web straps). At least two accidents occurred during this review period when forklift equipment was used in an unconventional but acceptable manner. Safe execution of these maneuvers requires experience and proficiency in both forklift operation and crane-related hoisting and rigging techniques.

The complexities associated with crane operations require highly trained personnel who generally gain proficiency through frequent repetition of H&R tasks. Forklift operation, while it does require training, appears significantly less complicated. Because use of this equipment is often incidental and not repetitive, personnel are generally not afforded the opportunity to gain proficiency. Personnel who use forklifts to perform warehousing tasks are an exception. In this environment the forklift is the principal tool, and operators generally receive significant training, perform repetitive tasks, and acquire proficiency. Probably for these reasons, fewer than 23 percent of forklift incidents were associated with warehousing activities.

The use of mobile cranes by subcontractors is expected to increase, heightening the need for effective oversight of subcontractors' safety performance.

Discussions with H&R experts within the DOE (Federal workers, contractors, and subcontractors) indicate that as production-related operations are curtailed and superseded with activities directed at waste management, environmental restoration, and facility dismantlement, the need for stationary or overhead cranes will be reduced, and mobile units will be in more demand. Mobile cranes owned and operated by subcontractors are often used to perform materials handling tasks of varying complexity, whereas overhead cranes are generally operated by contractors and are used to perform maneuvers that are relatively simple and often routine. Independent evaluations performed by the Office of Oversight, in addition to information reported into ORPS and the Department's Computerized Accident/Incident Reporting System (CAIRS), have highlighted deficiencies in oversight of subcontractor activities. Therefore, the additional risks posed as more H&R tasks involving cranes are performed by subcontractor personnel heightens the concern over H&R safety and the need for effective oversight of subcontractor performance. Information contained in ORPS does not explicitly and formally identify whether an H&R incident is associated with a contractor or subcontractor activity. While it was possible in this review to make this determination for some of the 131 incidents analyzed, it was not possible to resolve this issue for the entire sample.

Nearly half of all incidents involving equipment other than cranes and forklifts resulted in an accident.

"Other" H&R equipment (e.g., hoists, chainfalls, and block and tackle) are not for heavy duty use, as are cranes and forklifts; they are generally used to handle light loads that are not usually classified as critical lifts[5]. Like forklifts, this equipment is used incidentally to performing a task, and personnel operating it are commonly referred to as "incidental riggers." Personnel are usually not highly trained to operate this equipment, do not generally perform repetitive tasks, and are not afforded the opportunity to gain proficiency. Furthermore, the relatively lightweight, uncomplicated, and utilitarian characteristics of "Other" H&R equipment readily lend themselves to unconventional applications. Almost half of all incidents involving non-crane and non-forklift equipment resulted in an accident.

ROOT CAUSES OF HOISTING AND RIGGING INCIDENTS

Human error is the major cause of hoisting and rigging incidents.

Human error, whether directly associated with supervisors or equipment operators, is the principal cause of H&R incidents. Factors not related to human performance, such as equipment failure and weather, are responsible for only 6 percent of H&R incidents. Figure 2 presents information showing that management (35 percent) and personnel errors (33 percent) collectively account for 68 percent of all H&R incidents, as reported into ORPS.

Management shortcomings and workers' inattention to detail account for a large proportion of incidents.

Further analysis shows that deficient work planning (43 percent) and inadequate definition, dissemination, and enforcement of policy (24 percent) are responsible for two thirds of the incidents attributable to management deficiencies. Inattention to detail (56 percent) and not following procedures (28 percent) account for 84 percent of H&R incidents caused by personnel error. (See Figures 3 and 4, respectively.) Furthermore, inattention to detail is the most prevalent cause of all 131 H&R incidents, accounting for about one in every five incidents. Additionally, there are no indications that certain root causes are becoming less frequent over time, are being remedied, or are being replaced with other causal factors.

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[5]A lift is designated as critical if: 1) the load requires exceptional care in handling because of size, weight, close-tolerance installation, high susceptibility to damage, or other unusual factors, or 2) collision, upset, or dropping could result in either a) an impact, b) significant release of radioactive or other hazardous materials, or other undesirable conditions, c) undetectable damage that would jeopardize future operations or the safety of a facility, or d) damage that would result in unacceptable delay to schedule or other significant program impact. See U.S. Department of Energy Hoisting and Rigging Handbook, dated June 1995.

Table 2 provides information that can be used to support actions to reduce H&R incidents, based on an analysis of ORPS root cause categories. The data indicate that a generic remedy is not applicable to all H&R situations. For example, while inattention to detail — the single leading cause of all H&R incidents—is responsible for about one in every five crane and forklift incidents, it is associated with less than one in every ten incidents involving "Other" types of H&R equipment (e.g., hoists, chainfalls, block and tackle).

Work planning is a significant factor in non-forklift incidents, while the work environment has more effect on forklift incidents.

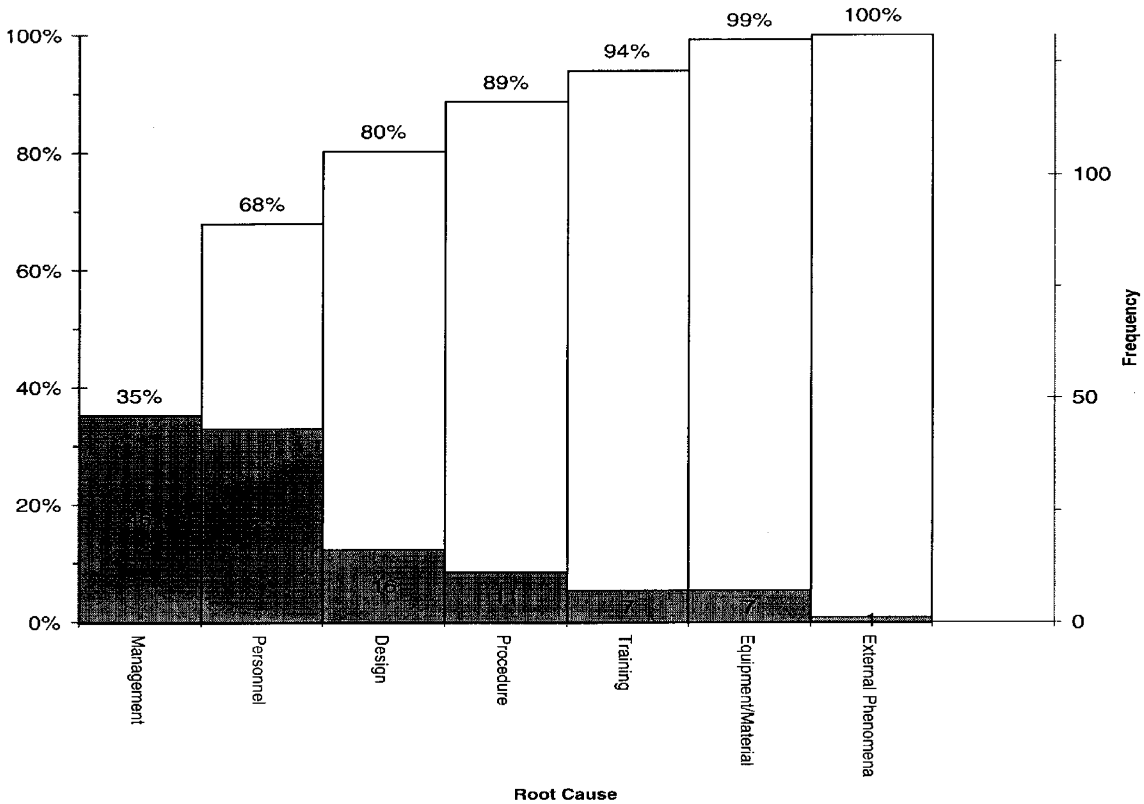
Work organization and planning require more attention in operations involving cranes and "Other" hoisting equipment than when forklifts are utilized. This is evident by the fact that inadequate work planning was the cause of 18 percent of all incidents involving cranes, 27 percent of the incidents involving "Other" hoisting (i.e., non-forklift) equipment, and only 3 percent of all forklift incidents. Similarly, the work environment (i.e., the characteristics of the area in which H&R equipment is operated) has a significantly greater influence on the frequency of forklift incidents than non-forklift incidents. As noted earlier, this is largely due to the mobility of forklifts and the increased likelihood of an incident when forklifts are used to transport loads over routes that are not protected from obstacles or other risks.

Materials handling activities that require the use of "Other" types of H&R equipment, including hoists, chainfalls, and block and tackle, are often initiated on an ad hoc basis and in response to an immediate need to perform a specific task. In these situations, the mechanics of the operation are generally not rigorously addressed, nor is the work well organized and planned. Approximately one in every five incidents involving this equipment is caused by defective engineering design or inadequate configuration of the equipment for the task being performed.

Training-related deficiencies were not identified as a major problem.

Surprisingly, training-related deficiencies were not identified as a significant problem. Procedure-related problems, including applying procedures incorrectly, defective or inadequate procedures, or procedures not used, are responsible for 18 and 20 percent of crane and forklift incidents, respectively. They were not found as causal factors for incidents involving "Other" equipment. Communication, lack of procedures, and defective or failed parts cause incidents with approximately equal frequency for all equipment type categories, although it is the greatest for "Other" equipment (e.g., hoists, chainfalls, block and tackle).

Figure 2. Major Root Causes of Hoisting and Rigging Incidents



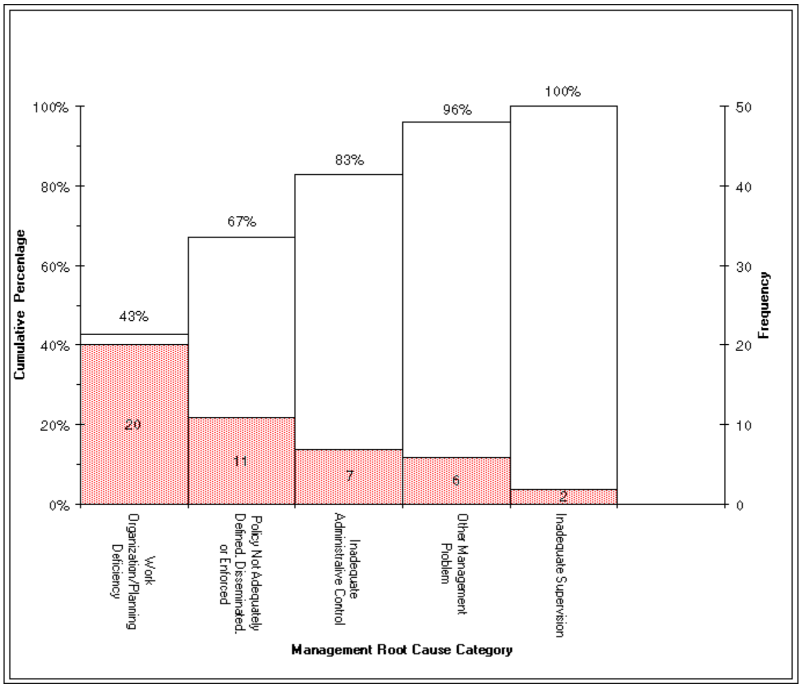


Figure 3. Management Root Cause Categories

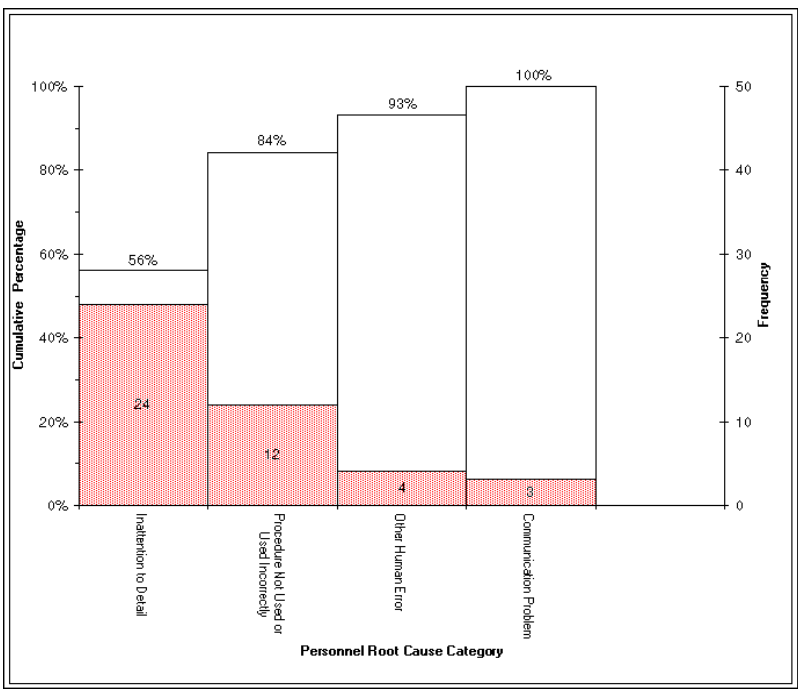


Figure 4. Personnel Root Cause Categories

Table 2. Root Cause of Hoisting and Rigging Incidents by Equipment Type*

Root Cause	Crane	Forklift	Other
Inattention to Detail	20%	23%	8%
Work Organization and Planning	18%	3%	27%
Procedure Not Used or Used Incorrectly	9%	15%	0%
Policy Not Adequately Defined, Disseminated, or Enforced	9%	10%	4%
Inadequate or Defective Design	5%	5%	19%
Defective or Inadequate Procedure	9%	5%	0%
Inadequate Administrative Control	9%	0%	4%
Defective or Failed Part	5%	5%	8%
Other Management Problem	3%	3%	12%
Other Human Error	3%	3%	0%
Inadequate Work Environment	0%	10%	0%
Lack of Procedure	2%	3%	4%
Insufficient Refresher Training	3%	3%	0%
Insufficient Practice or Hands-On Experience	5%	0%	0%
Communication Problem	2%	3%	4%
Inadequate Supervision	0%	3%	4%
Error in Equipment or Materials Selection	0%	3%	4%
Weather	0%	3%	0%
No Training Provided	0%	0%	4%

*Rounded to the nearest whole number.

4.0 CONCLUSIONS

This section presents the major conclusions based on the study results presented in Section 3. These conclusions are applicable Department-wide, and provide a foundation for candidate future actions to improve H&R safety performance.

Management attention is needed to improve the safety of hoisting and rigging operations.

- Despite numerous incidents, accidents, and the lessons-learned workshop, there has been no significant improvement in H&R activities. The manner in which H&R tasks are performed and the associated adverse consequences are consistent with an unchanged process that exhibits expected variations in safety performance. While additional independent oversight may not alleviate the current situation, line management can improve safety by implementing specific actions to change the process by which H&R operations are performed and overseen by line management.

Specific corrective actions depend on the type of equipment being used.

- Root causes of H&R incidents display a strong relationship to the type of equipment used. Thus, management may consider formulating equipment-specific corrective actions to improve H&R safety performance.

- H&R equipment items used incidentally, such as forklifts, are associated with a large proportion of accidents. Effective initiatives by management to address these operations and type of usage will realize significant improvement in H&R safety performance.

- As the Department transitions from production-oriented operations to environmental restoration, greater use of mobile cranes operated by subcontractors can be expected. This situation suggests close evaluation and monitoring by management to limit or prevent H&R incidents and accidents.

APPENDIX A
ANALYTICAL AND STATISTICAL METHODS

INFERENTIAL STRENGTH OF SAMPLE DATA

A narrative search, using the search string rigg@+hoist@+crane+forklift+sling, in accordance with the *ORPS User's Manual*, was performed on the entire ORPS database covering reports from January 1, 1988, to March 31, 1996. This process yielded 1,187 occurrence reports relating to H&R; 491 of these were associated with the 30-month period analyzed. Applying the criteria identified in Section 2 of this report to these 491 reports resulted in 131 relevant H&R incidents. Assuming that there were no events that had a significant impact on the reporting level of H&R occurrences throughout this period, proportional analysis can be applied as follows:

$131/491=X/1187$

$X=317$

It follows then that of the 1,187 occurrence reports in ORPS that relate to H&R, 317 fulfill the aforementioned criteria and represent the population of relevant incidents. Therefore, the 131 incidents analyzed represent approximately 41 percent of the total population (i.e., 131/ 317 = .41). This provides a basis for performing an extrapolation and making inferences to the entire population of relevant H&R incidents (317) on the results from analyzing the sample (131).

RANDOMNESS OF VARIATION IN SAFETY PERFORMANCE

A statistical test performed on the data suggests that there is only a 5 percent chance that the variation in the number of incidents over the 30- month period is due entirely to random influences[1].One plausible explanation for this cyclical phenomenon, therefore, is the factors associated with human learning and short term memory. Generally, immediately following an incident there is a short but pronounced period when individuals are most conscious of avoiding the same or similar mistakes; during this time, they demonstrate improved safety performance. Over time, however, without reinforcement (e.g., training, lessons learned, reminders) the sense of urgency and attentiveness generated by the incident declines, and poor safety habits resurface. Eventually, an incident occurs and the cycle repeats itself.

CONSTRUCTION OF CONFIDENCE LIMITS

Inferences about the population of relevant H&R incidents (317) contained in ORPS can be made based on the sample (131) with a conservative degree of confidence. Table A-1 contains a summary of confidence limits for significant sample statistics, corresponding to the .95 confidence coefficient.

[1]The theory of runs from H.T. Davis, *The Analysis of Economic Time Series,* pp. 164-170, was applied to test the null hypothesis that the cyclical variation is random. A chi-square test was significant at the .05 level, indicating that there is a 95 percent probability that the cyclical variation is not due to chance.

For example, the table shows that while 50 percent of the 131 sample incidents analyzed involved cranes, one can be 95 percent confident that the proportion for the total population of 317 relevant incidents contained in ORPS lies in the interval between 41 and 59 percent. Similarly, one is 95 percent confident that between 27 and 43 percent of all relevant H&R incidents contained in ORPS are due to management deficiencies, and that between 11 and 25 percent are caused by inattention to detail. Establishing confidence limits puts the utility of the sample results in perspective. Confidence limits help highlight general conclusions and, more importantly, aid in the selection process used to implement discrete recommendations. The established confidence interval around the H&R sample statistics can be used to determine best, median, and worst case scenarios when quantifying impacts of alternative safety improvement strategies — e.g., performing benefit-cost and cost-effectiveness analyses. Proposed actions to limit or prevent H&R incidents and accidents are generally analyzed with respect to expected outcomes that are consistent with established confidence limits.

Table A-1. Approximate 95 Percent Confidence Limits for Selected Sample Statistics

Parameter	Sample Statistic	Lower Confidence Limit*	Upper Confidence Limit*
Crane Incidents as a Percent of Total	50%	41%	59%
Forklift Incidents as a Percent of Total	31%	23%	39%
Crane Accidents as a Percent of Total	51%	42%	60%
Forklift Accidents as a Percent of Total	38%	30%	46%
Management Deficiency	35%	27%	43%
Personnel Error	33%	25%	41%
Inattention to Detail	18%	11%	25%
Work Organization and Planning	15%	9%	21%
Procedure Not Used or Used Incorrectly	9%	4%	14%
Policy Not Adequately Defined, Disseminated, or Enforced	8%	3%	13%

*Calculated using the normal approximation to the binomial distribution.

APPENDIX B
TEAM COMPOSITION

Deputy Assistant Secretary, Oversight: Glenn S. Podonsky

Associate Deputy Assistant Secretary: Neal Goldenberg

Director, Office of Oversight Planning and Analysis: Rebecca F. Smith

Deputy Director, Office of Oversight Planning and Analysis: Frank B. Russo

Analyst: David M. Berkey

Appendix D

Crane Accidents 1997 - 1999
California Department of Industrial Relations

May 2000

INTRODUCTION

This appendix includes an Occupational Safety and Health Administration (OSHA) report concerning crane accidents from 1997 through 1999. Data for the OSHA Crane Report was gathered from Federal OSHA's Office of Management Data Services (OMDS) Website. Unfortunately, the findings that are made in the report are gross failures, and were not normalized by load weight, crane capacity, type of industry, or the number of failures per demand.

Several observations of the OSHA report are similar to this and other crane operating reports.

(1) The number of crane accidents occurring during construction activities was about the same as crane accidents that occurred during non-construction activities. Of the 158 crane accidents, 80 accidents occurred during non-construction work and 78 during construction-related work. It is assumed that “non-construction” crane accidents included general routine maintenance or industrial activities involving load movements.

(2) Crane accidents were dominated by mobile cranes. Of the 158 crane accidents, mobile cranes accounted for 73 percent of the accidents, bridge cranes 16 percent, gantry cranes 3 percent, tower cranes 3 percent, and ship cranes 1 percent. There were 7 crane accidents (4 percent) where the type of crane involved was not known.

(3) More accidents occurred in the private sector. Of the 158 crane accidents, 150 accidents involved private sector entities and 8 involved public sector entities. Of the 8 public sector cases, 7 resulted in serious injuries.

(4) Public sector crane accidents were dominated by mobile cranes. All 8 of the public sector cases involved a mobile crane.

CRANE ACCIDENTS 1997 - 1999:

A REPORT OF THE CRANE UNIT OF THE DIVISION OF OCCUPATIONAL SAFETY AND HEALTH

Prepared by Philip Yow
Associate Cal/OSHA Engineer

23 May 2000

Ray Rooth, Principal Safety Engineer
Research and Standards Safety Unit

and

Ken Fry, Senior Cal/OSHA Engineer
Crane Certifier Accreditation Unit

Division of Occupational Safety and Health
California Department of Industrial Relations

I. Data Sources

A. Data for the Crane Report was gathered from Federal OSHA's Office of Management Data Services (OMDS) Website. Searches were made on the OMDS website by:

- Various keywords, e.g. "crane," "mobile crane," "hydraulic crane," "load"
- 2, 3, and 4 Digit Standard Industrial Classification (SIC) Codes, e.g., 15xx, 16xx, 17xx
- Establishments – “Crane XXX”

B. Data was also gathered from Micro-to-Host Reports from the Integrated Management Information System (IMIS). The following requests were made:

- All Accidents Reports (Form 36)[1] from 1/1/97-12/31/99
- All Inspections with Optional Information “S-10 Cranes”
- All Citations involving 8 CCR 4999 and 8 CCR 2946

II. Total Number of Crane Accidents

From 1 January 1997 through 31 December 1999, the Division of Occupational Safety and Health learned of, or had reported to it, a total of 158 accidents involving a crane.

Over the three-year period from 1997 through 1999, at least one crane accident has occurred in each month of the three year period.

III. Types of Cranes Involved in Accidents

The types of cranes involved in the 158 accidents are as follows (N=158):

Crane Type	Count	Percentage
Mobile Cranes	115	73%
Bridge Cranes	26	16%
Gantry Cranes	5	3%
Tower Cranes	4	3%
Ship Cranes	1	1%
Not Determined[1]	7	4%

[1] 8 CCR Section 342(a) requires that “serious injury or illness, or death of an employee occurring in place of employment or in connection with any employment” be reported to the Division. Because only accidents with serious employee injuries, or death, are reported, the exact number of crane accidents is not known. See attached Excel Spreadsheet for all reported crane accidents for the period 1997-99.

IV. Crane Operator and Non-Crane Operator Injuries

A. Total Injuries, Serious and Fatal, By Type of Worker

1. Crane Operator -- One Fatal Injury and 23 Non-Fatal Injuries

   While fourteen of the operators injured were bridge crane operators, the one fatality was a mobile crane operator.

2. Non-Crane Operators -- 12 Fatal Injuries and 79 Non-Fatal Injuries

   a. Ninety-one non-crane operators were injured in crane accidents. Of the 91 crane accident-related injuries, 72 of these accidents involved mobile cranes. These non-crane operators include occupations such as mechanics, oilers, ironworkers, riggers, and stevedores.

   b. In this category, 12 involved fatal injuries 8 of which involved non-crane operators who were engaged in work in the vicinity of mobile cranes.

3. Of the total of 13 fatalities for crane operators and non-crane operators, four (4) were the result of falling loads. All of the falling load fatalities involved a mobile crane. There were 3 fatalities from the 14 electrical contact accidents. Two of the electrical contact fatalities involved mobile cranes.

   Although there were 35 mobile crane “tip-over” accidents, there was only one fatality in all tip-over accidents when a worker was killed when a crane tipped over onto him.

V. Private vs. Public Sector Crane Accidents

Of the 158 crane accidents, 150 accidents involved private sector entities and 8 involved public sector entities. Of the 8 public sector cases, 7 resulted in serious injuries. All 8 of the public sector cases involved a mobile crane.

VI. Construction vs. Non-construction Crane Accidents

Of the 158 crane accidents, 80 accidents occurred in non-construction work and 78 in construction-related work.

VII. Accident Causation

A. Most Frequent Causes: All Crane Types (N-158) & Mobile Cranes (N-115)

	All Crane Types	Mobile Cranes
1.Instability	67	49
a. Unsecured Load	34	6
b. Load Capacity Exceeded	0	29
c. Ground not level/too soft	0	4
2.Lack of Communication	32	24
3.Electrical Contact	13	10
4.Misc. in 14 Categories	46	32

B. Instability, Lack of Communication and Other Causal Factors

1. Instability

Instability accidents for mobile cranes generally resulted in either the crane tipping over, or the load falling off the hook or slings. Instability accidents were further broken down into separate categories.

2. Lack of Communication

Lack of communication was another major cause of accidents because the point of operation is usually some distance from the crane's operator station or not in full and direct view of the operator in operations involving mobile cranes. Seventy-five percent of accidents caused by both "lack of communication" and "electrical contact" involved mobile cranes.

3. Lack of Training

Although "Lack of Training" did not rank very high as a primary cause, it would have been ranked within the top three if a secondary cause were listed.

[1] In seven cases, there was insufficient information available to determine the specific type of crane involved.

Appendix E

U.S. Navy Crane Operating Experience

INTRODUCTION

This appendix provides information on U.S. Navy crane operating experience data. Operating experience obtained from the Navy has been used by industries utilizing cranes, to reduce the risk and financial impact of crane accidents. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” (published in 1980) relied heavily on U.S. Navy crane operating experience. The Navy crane data used in NUREG-0612 included summaries of 466 crane events covering a period from February 1974 to October 1977. An exact accounting of the number of lifts per year made by each crane was not available from the Navy. Estimates were made of the number of lifts, and of the number of load drops due to changes in the number of facilities and vessels covered in the reporting system. The Navy crane data included summaries of 66 crane events covering a period from December 1995 to May 1999. An exact accounting of the number of lifts per year made by each crane was not available from the Navy. Once again, estimates were made of the number of lifts, since this information was not available.

Table E1, “Reported U.S. Navy crane events (1995-1999),” provides a listing of the 66 Navy crane events. Each crane event is listed by crane type, accident type, accident cause, responsible group, function being performed at the time of the event, and crane operating mode. A breakdown is also provided showing the end result of the crane event and its cause. Abbreviations used in Table E1 are shown on Table E2, “U.S. Navy crane data abbreviations.”

As shown in Table E1, human factors or human errors are the leading causes of Navy crane issues. This would include the categories of Improper Operation (IO), Improper Rigging (IO), and Procedure Failure (PROC), which accounted for approximately 88 percent of crane issues. Those crane issues related to crane equipment failures accounted for approximately 5 percent of crane issues. These findings are similar to this and other studies of crane operating experience.

Table E1: Reported U.S. Navy crane events (1995-1999)

Report#	Date	Crane	Accident Type	Accident Cause	Responsible Group	Function	Operating Mode		Issue			
95001	5/9/95	BNS	DC	IR	R	H	OP		Damaged Crane	Cause	Percent	# Reports
95001	12/5/95	BNS	DROP	IR	R	H	OP			IO	50.0	9
95002	8/30/95	BNS	CC	TC	M	T	MAIN			IR	27.8	5
96001	2/8/96	ONS	DC	IO	UNK	H	OP			PROC	22.2	4
96002	7/1/96	ONS	PI	IR	SHOP	H	OP					
96002	9/9/96	BNS	CC	II	CONT	T	OP		Crane Collision	Cause	Percent	# Reports
96004	4/25/96	BS	PI	IR	R	NA	OP			IO	45.5	5
96005	7/13/96	BS	DL	IR	R	H	OP			PROC	18.2	2
96010	9/25/96	BS	DL	IR	R	H	OP			Others	36.4	4
96013	7/9/96	BS	DC	IR	R	H	OP					
96014	12/3/96	BS	UL	IO	O	L	ODCL		Load Collision	Cause	Percent	# Reports
96017	9/6/96	BS	DL	EQ	EC	H	OP			IO	55.6	5
96028	8/9/96	BS	DC	IO	O	HT	OP			IR	22.2	2
96041	11/8/96	BS	LC	IR	R	H	OP			PROC	11.1	1
97001	11/4/97	BNS	CC	IO	O	B	OP			VISI	11.1	1
97001	1/8/97	BS	PI	IR	R	IDLE	OP					
97001	2/5/97	BNS	OVER	IO	O	H	OP		Overload	Cause	Percent	# Reports
97001	2/10/97	BNS	PI	IO	O	H	OP			IO	25.0	2
97001	4/29/97	BNS	DROP	IR	R	H	OP			IR	37.5	3
97002	4/18/97	BNS	PI	PROC	WELD	H	OP			PROC	37.5	3
97003	10/30/97	BNS	SHOCK	IR	FWORK	H	OP					
97004	2/27/97	BS	LC	VISI	R	H	OP		Personnel Injury	Cause	Percent	# Reports
97008	10/1/97	BNS	CC	IO	O	T	OP			IO	20.0	1
97008	2/26/97	BS	CC	PROC	EC	T	OP			IR	60.0	3
97009	11/19/97	BNS	OVER	PROC	MG	H	OP			PROC	20.0	1
97010	10/1/97	BNS	CC	IO	UNK	B	OP					
97013	12/1/97	BS	DC	IO	O	UNK	OP		Dropped Load	Cause	Percent	# Reports
97014	6/2/97	BS	OVER	PROC	MG	H	OP			EQ	40.0	2
97016	4/30/97	BS	DROP	EQ	R	ROT	OP			IR	60.0	3
98001	4/28/98	BNS	DC	IO	O	TROL	OP					
98001	2/26/98	BNS	DC	PROC	MECH	L	MAIN		Two-Blocking	Cause	Percent	# Reports
98001	1/13/98	BNS	DC	IO	O	H	OP			IO	66.7	2
98001	11/4/98	BNS	DROP	EQ	R	H	OP			PROC	33.3	1
98001	1/6/98	BS	DROP	IR	R	L	OP					
98002	2/18/98	BS	DR	IR	R	H	OP		Damaged Load	Cause	Percent	# Reports
98002	3/13/98	GNS	DC	IO	MG	N/A	OP			EQ	33.3	1
98004	8/4/98	GNS	DC	PROC	ISP	H	OP			IR	66.7	2
98004	5/5/98	GNS	CC	IO	O	T	OP					
98004	2/2/98	BNS	DC	IO	O	B	MAIN		Damaged Rigging	Cause	Percent	# Reports
98004	11/18/98	BNS	DC	IR	R	H	OP			IR	1.0	1
98004	2/11/98	BNS	LC	IO	O	L	OP					
98005	1/8/98	BNS	TB	IO	CONT	H	UNK		Uncontrolled Lowering	Cause	Percent	# Reports
98006	12/9/98	GNS	CC	VISI	R	T	OP			IO	1.0	1
98006	4/10/98	BNS	DC	IR	R	L	OP					
98007	5/15/98	BS	LC	IO	R	B	OP		Shock	Cause	Percent	# Reports
98008	7/29/98	BNS	OVER	PROC	MG	H	OP			IR	1.0	1
98008	6/26/98	BS	TB	PROC	M	H	MAIN					
98009	7/21/98	BS	CC	IO	O	B	OP		Other	Cause	Percent	# Reports
98010	7/2/98	BS	LC	IO	O	L	OP			PROC	1.0	1
98010	12/22/98	BS	OVER	IO	R	H	OP					
98013	8/28/98	BS	OVER	IR	R	H	OP					
98017	11/6/98	BS	OTHER	PROC	CONT	T	OP					
98018	12/11/98	BS	OVER	IR	R	H	OP					
98029	12/14/98	BS	CC	PROC	O	T	OP					
99001	10/30/98	BNS	LC	IO	O	H	OP					
99001	3/30/99	BNS	LC	IO	R	H	OP					
99002	4/12/99	BNS	DC	IR	IR	H	OP					
99003	2/22/99	MONO	TB	IO	O	H	OP					
99003	4/13/99	BS	OVER	IR	R	H	OP		Summary by Cause	Cause	Percent	# Reports
99003	10/15/98	BNS	DC	IO	O	T	OP			IO	37.9	25
99004	10/15/98	BNS	DC	IO	O	H	OP			IR	30.3	20
99005	3/9/99	BS	LC	PROC	R	HOLD	TEST			PROC	19.7	13
99007	1/29/99	BNS	DC	PROC	M	T	MAIN			EQ	4.5	3
99008	3/30/99	BS	LC	IR	R	T	OP			Misc	7.6	5
99008	3/20/99	BNS	DC	PROC	MG	T	OP			Total	100.0	66
99013	5/8/99	BS	CC	COMM	ISP	TROL	ISP					

Table E1 abbreviations are shown in Table E2

Table E2: U.S. Navy crane data abbreviations

B	Bridge movement	M	Maintenance (personnel)
BS	Bridge crane	MAIN	Maintenance (mode of operation)
BNS	Bridge crane	MECH	Mechanic (personnel)
CC	Crane collision	MG	Management/supervision
COMM	Communication problem	MONO	Monorail crane
CONT	Contractor	O	Operator (personnel)
DC	Damaged crane	ODCL	Surveillance
DL	Damaged load	ONS	Overhead crane
DR	Damaged rigging	OP	Operation (mode of operation)
DROP	Load drop	OTHER	Not directly related to a crane
EC	Personnel other than shop personnel	PI	Personal injury
EQ	Equipment failure	PROC	Procedure failure
FWORK	Foundry worker	R	Rigger
GNS	Gantry crane	ROT	Load rotation
H	Hoisting	Shock	Shock load (similar to drop)
HOLD	Holding	T	Crane travel/movement
HT	Hoisting/travel	TB	Two-blocked
IDLE	Idle	TC	Track condition
II	Improper installation	TEST	Testing (mode of operation)
IO	Improper operation	TROL	Trolley/bridge movement
IR	Improper rigging	UL	Uncontrolled lowering
ISP	Inspection/maintenance (mode of operation)	UNK	Unknown
L	Lowering	VISI	Inadequate visibility
LC	Load collision	WELD	Welder (personnel)

Appendix F

Load Drop Calculations Involving Heavy Loads at U.S. Nuclear Power Plants

INTRODUCTION

This appendix includes a partial listing of load drop calculations obtained from each facility that was visited. A review of licensee provided load drop calculations indicated that calculational methodologies and assumptions varied greatly from licensee to licensee, producing radically different end results or consequences. Table F1, “Heavy load drop calculations,” provides a partial listing of load drop calculations obtained from the facilities that were visited. Table F2, “Heavy load calculation abbreviations,” provides a listing of abbreviations that are used in Table F1. Heights of load drops, plant locations for postulated load drops, contact areas at impact, materials property values, and weights of loads varied greatly. The Oyster Creek calculation for a drop of a fuel cask weighing 41metric tons (45 tons) over a 41 cm (16 inches) thick reinforced concrete slab was the most restrictive, with an allowable drop height of 7 cm (2.77 inches). Other calculations listed in Table F2 indicate acceptable results with much heavier loads, and with greater drop heights, than those used in the Oyster Creek example. Some facilities performed load drop calculations using equations that were intended for ballistic type situations meant for high velocity and low mass, rather than low velocity and high mass which would be typical for heavy load drop scenarios. Each licensee used load drop calculations to determine transport height restrictions in their heavy load procedures. These restrictions should be based on conservative and consistent engineering analyses.

Table F1: Heavy load drop calculations

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Grand Gulf	8/15/78 Bechtel	Drywell head	56 (61.5 tons)	1.5 m (5 feet) (in air)	Refueling floor; 23 cm (9 inches) thick RC slab on 7.6 cm (3 inches) decking (non-composite), slab supported on W36x300 beams @ 1.9 m (76 inches) spacing	5.5 m/sec (17.9 feet/sec)	Used an equation for penetration of 30.5 cm (12 inches) diameter missiles. 100 percent of flange will contact the floor.	- Depth of penetration 7 cm (2.8 inches) - 23 cm (9 inches) thick RC slab μ = 6.9 - W36x300 μ = 5.9
Grand Gulf	8/17/78 Bechtel	Drywell head	56 (61.5 tons)	9.2 m (30 feet) (in air)	Reactor well; 3.8 cm (1.5 inches) wide sleeve, radius of 4.9 m (16.1 feet)	13.4 m/sec (43.9 feet/sec)	Drywell head hits the sleeve	- Drywell head crushes the sleeve, and continues downward, but does not compromise the integrity of the RPV
Grand Gulf	8/16/78 Bechtel	RPV head	106 (117 tons)	1.5 m (5 feet) (in air)	Refueling floor, 1.2 m (4 feet) thick RC	5.5 m/sec (17.9 feet/sec)	100 percent of flange will contact the floor	-Depth of penetration 11.2 cm (4.4 inches) -For simple support, μ=9 - For fixed support, μ<1
Grand Gulf	4/4/78 Bechtel	Steam separator	62 (68 tons)	5.2 m (17 feet) (in water)	Spent fuel pool; Steam separator area, 1.3 m (52 inches) thick slab with 0.6 cm (1/4 inch) liner plate	6.6 m/sec (21.5 feet/sec)	Steam separator falls in water	-Assuming a 0.6 cm (1/4 inch) thick plate, the depth of penetration 1.8 cm (0.7 inch) (unsatisfactory) -Assuming a 1.3 m (52 inches) thick concrete slab, depth of penetration 15.7 cm (6.2 inches) -Assuming an interface forcing function, depth of penetration 6.6 cm (2.6 inches) -Using a structural response and ratioing, the slab response will not exceed the acceptable ductility ratio of 10
Grand Gulf	7/18/78 Bechtel	Steam dryer	36 (40 tons)	7 m (23 feet) (in air)	Dryer storage area, 12.7 cm (5 inches) thick slab with 0.6 cm (1/4 inch) thick liner plate	11.7 m/sec (38.5 feet/sec)	For the 0.6 cm (1/4 inch) thick liner plate, the equation appears to spread out the load over an entire cylinder with a diameter of 6 m (238 inches) (same for the slab) as opposed to an annulus.	- Assuming a 0.6 cm (1/4 inch) thick plate, the depth of penetration 0.23 cm (0.09 inch) - Assuming a 1.3 m (52 inches) thick concrete slab, depth of penetration 13.7 cm (5.4 inches) - μ <5.3

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oyster Creek	10/29/99 EQE	Fuel cask	41 (45 tons)	15 cm (6 inches) (in air)	Refueling floor; At the center of beam 5B27; slab thickness 41 cm (16 inches); beam width 91 cm (36 inches), beam depth 76 cm (30 inches); various rebar 20 to 38 cm (8 to15 inches)	1.7 m/sec (5.7 feet/sec)	ACI 349-97	- Allowable drop height 17.8 cm (7.01 inches)
Oyster Creek	10/26/99 EQE	Fuel cask	41 (45 tons)	9.8 cm (3.85 inches) (in air)	Refueling floor; center drop on slab 5S10; slab span N/S 7 m (23 feet), 7.6 cm (3 inches) x E/W 6.1 m (20 feet), 23 cm (9 inches); slab thickness 41 cm (16 inches); #6 rebar @17.8 cm (7 inches) and 45.7 (18 inches) centers, and #8 rebar @ 15, 20, and 23 cm (6, 8, and 9 inches) centers	1.4 m/sec (4.55 feet/sec)	ACI 349-97	- Allowable drop height 9.8 cm (3.85 inches)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oyster Creek	10/26/99 EQE	Fuel cask	41 (45 tons)	7 cm (2.77 inches) (in air)	Refueling floor; Drop on slab 5S10 adjacent to beam 5B27; slab span N/S 7 m (23 feet), 7.6 cm (3 inches) by E/W 6.1 m (20 feet), 23 cm (9 inches); slab thickness 41 cm (16 inches); #6 rebar @18 cm (7 inches) and 46 cm (18 inches) centers, and #8 rebar @ 15, 20, and 23 cm (6, 8, and 9 inches) centers	1.2 m/sec (3.86 feet/sec)	ACI 349-97	- Allowable drop height 7 cm (2.77 inches)
Oyster Creek	10/26/99 EQE	Fuel cask	41 (45 tons)	29.4 cm (11.58 inches) (in air)	Refueling floor; Drop on slab 5S14 adjacent to beam 5B39; similar to slab 5S10 but 66 cm (26 inches) thick slab	2.4 m/sec (7.88 feet/sec)	ACI 349-97	- Allowable drop height 29.4 cm (11.58 inches)
Oyster Creek	10/26/99 EQE	Fuel cask	41 (45 tons)	15 cm (6 inches) (in air)	Refueling floor; Drop on east wall of spent fuel pool; the wall is 1.8 m (6 feet) thick and extends from the 36 m (119 feet) level to the 22 m (72 feet) level	1.7 m/sec (5.7 feet/sec)	Analyzed as a hard object striking a hard target; the drop would occur between columns C5 and C6 and between beam 5B21 and 5B19, and slab 5S14	- If kinetic energy of drop is set equal to the strain energy, the allowable drop height would be 1.26 m (49.6 inches) - If load is dropped directly on C6, the allowable drop height would be 1.24 m (49 inches)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Palo Verde	6/4/80 Bechtel	Fuel cask	114 (125 tons)	3.7 m (12 feet) (in air)	Drop from level 38 m (124.5 feet) to the decontamination pit (about 3.7 m [12 feet])	8.5 m/sec (27.8 feet/sec)	Assumes that the cask hits the floor exactly flat; Ductility ratio of 30 acceptable	- Thickness required to preclude spalling 1.82 m (71.56 inches); slab defection 0.16 cm (0.063 inches); ductility ratio calculated to be 22.84
Palo Verde	6/4/80 Bechtel	Fuel cask	114 (125 tons)	9.2 m (30 feet) (in air)	Drop from the top of the spent fuel pool to the bottom of the cask loading pit; target slab is 2.4 m (93 inches) thick	13.7 m/sec (45.0 feet/sec)	Ductility ratio of 30 acceptable	- Ductility ratio of 6.01 calculated, 30 is acceptable - using a different soil reaction, ductility ratio calculated to be 9.67, 30 is acceptable
Palo Verde	6/23/80 Bechtel	Fuel cask	114 (125 tons)	30.5 cm (1 foot)	Drop from top of spent fuel pool to the decontamination pit and then deflects (rotation strike on wall) to the east wall of the pit	Striking velocity on the wall 4.9 m/sec (16.1 feet/sec)	Ductility ratio of 30 acceptable	- Calculated ductility ratio 47.09, 20 (average of beam at 10, and slab at 30) - For this situation, an energy absorbing pad was required
Brown's Ferry	1/14/72 TVA	Fuel cask	91 (100 tons)	0.9 m (3 feet)	Drop on hypothetical 45.7 cm (18 inches) thick RC slab	4.2 m/sec (13.9 feet/sec)	NAVDOCKS (page 51); Cask lands flat on 16 fins, evenly distributed (0.38 square meters [4.124 square feet])	- Depth of penetration 2.7 cm (0.0892 foot)
Brown's Ferry	1/17/72 TVA	Fuel cask	91 (100 tons)	0.9 m (3 feet)	Drop on hypothetical 45.7 cm (18 inches) thick RC slab	4.2 m/sec (13.9 feet/sec)	Compares energy absorbed to the energy the system can ultimately absorb	- Reaches 77 percent of maximum energy
Brown's Ferry	1/18/72 TVA	Fuel cask	91 (100 tons)	0.9 m (3 feet)	Drop on 45.7 cm (18 inches) thick slab near supports	4.2 m/sec (13.9 feet/sec)	After punching through in the area immediately adjacent to the slab support, the structural system will form two effective cantilever beams with three plastic hinges	- Punch through will occur near the column and beams in an arc, but will not go through the slab
Brown's Ferry	1/27/72 TVA	Fuel cask	91 (100 tons)	15 cm (6 inches)	Drop on 0.91 m (36 inches) thick slab	1.7 m/sec (5.7 feet/sec)	Uses a modified Petry formula for penetration	- Penetration calculated to be 0.46 cm (0.015 foot)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/30/84 Bechtel (1)	Drywell head	95 (104 tons)	0.91 m (3 feet)	Tilted drop on refueling floor, RC slab 61 cm (24 inches) thick, #9 rebar @ 20 cm (8 inches) centers (T&B)	4.2 m/sec (13.9 feet/sec)	Capacity of slab based on yield-line theory, simple span, elasto-plastic design; Does not appear to account for kinetic energy absorption over a small area; Tilted drop case, strikes over 40 degrees of circumference	- Punching shear capacity appears to be high (1.66 MPa [240 psi]) - Calculated punching shear appears to be low (0.81 MPa [117psi]) - Compressive strength of concrete appears to be high - Modulus for concrete appears to be high - μ = 0.8 , allowable 10 (over concrete, Zones A&B) - $\mu <$ 8.72, allowable 8.72 (over W36 beam, Zones A&B) - μ = 7.5, allowable 8.72 (over two W36 beams, Zones A&B) - $\mu <$ 1.0 (over concrete, Zone C) - $\mu <$ 12, allowable 20 (over W24) - μ = 10, allowable 12 (over two beams W24)
Limerick	4/24/84 Bechtel (2)	Drywell head	95 (104 tons)	0.91 m (3 feet)	Flat drop on refueling floor	4.2 m/sec(13.9 feet/sec)	Drywell head lands completely flat on the refueling floor	- Flat drop case shows a greater force on the floor than does the tilted case above - μ = 1.8, allowable 10 (over concrete zone A&B) - μ = 1.5, 8.72 allowable (over W36 beam, Zones A&B) - μ = 1.4 (over concrete, Zone C) - μ = 2, allowable 12 (over W24, zone C)
Limerick	4/26/84 Bechtel (3)	RPV Head	84 (92 tons)	0.91 m (3 feet)	Flat drop on refueling floor	4.2 m/sec (13.9 feet/sec)	RPV head lands completely flat on the refueling floor	- μ = 1.8, allowable 10 (over concrete, Zones A&B) - $\mu <$3 (over W36, zones A&B) - μ = 1.3, allowable 10 (over concrete, Zone C) - $\mu <$5 (over W24, zone C)
Limerick	4/26/84 Bechtel (4)	RPV Head	84 (92 tons)	0.91 m (3 feet)	Tilted drop on refueling floor	4.2 m/sec (13.9 feet/sec)	RPV head lands tilted	- Flat drop case shows a greater force on the floor than does the tilted case above - μ = 1.0 (over concrete, Zones A&B) - μ = 5.5 (over two beams, W36, zones A&B) - $\mu <$1.0 (over concrete, Zone C) - $\mu >$ 100 (over two beams, W24, Zone C)
Limerick	4/26/84 Bechtel (5)	RPV Head	84 (92 tons)	0.61 m (2 feet)	Tilted drop on refueling floor	3.5 m/sec (11.4 feet/sec)	RPV head lands tilted	- μ ~ 20 (over W24, Zone C) - Drop height was changed from 0.9 m (3 feet) to 0.61 m (2 feet) to get a lower value for μ

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/26/84 Bechtel (6)	Shield Plugs	11 (12 tons)	0.91 (3 feet)	Flat drop on refueling floor	4.2 m/sec (13.9 feet/sec)	Flat drop calculated for over W36, 61 cm (24 inches) thick concrete, and W24	- Flat drop force for the 11 metric ton (12 ton) plugs was calculated to be greater than the tilted drop of the drywell head at 95 metric tons (104 tons) - μ = 1.5, allowable 10 (over concrete, Zones A&B) - $\mu \lll 1$ (over W36, zones A&B) - μ =1.5, allowable 10 (over concrete, Zone C) - μ =2.4, allowable 10 (over W24, zone C)
Limerick	4/26/84 Bechtel (7)	Stoplog	54 (59 tons)	0.91 (3 feet)	Flat drop on refueling floor	4.2 m/sec (13.9 feet/sec)	Flat drop; Contact area 7 square meters (75 square feet)	- μ = 2, allowable 10 (over concrete, Zones A&B) - μ = 1.08, allowable 8.72 (over W36, zones A&B) - $\mu \prec 2.5$, allowable 10 (over concrete, Zone C) - μ = 1.53, allowable 10 (over W24, zone C)
Limerick	4/26/84 Bechtel (8)	Stoplog	54 (59 tons)	0.91 (3 feet)	Tilted drop (45 degrees) on refueling floor	4.2 m/sec (13.9 feet/sec)	Tilted drop; Contact area 0.23 square meters (2.5 square feet)	- Per an unreferenced equation, spalling of concrete will occur at a drop height of approximately 10 cm (4 inches) - μ = 0.6, allowable 10 (over concrete, Zones A&B) - Punching shear capacity appears to be high (1.66 MPa [240 psi] from page 12 of calculation) - Calculated punching shear appears to be low (1.2 MPa [173 psi]) - μ = 4, allowable 8.72 (over W36, zones A&B) - μ = .4, allowable 10 (over concrete, Zone C) - μ = 100, allowable 12 (over W24, zone C)
Limerick	4/26/84 Bechtel (9)	Stoplog	54 (59 tons)	53 cm (21 inches)	RC slab 61 cm (24 inches) thick refueling floor	3.2 m/sec (10.6 feet/sec)	Tilted drop; Contact area 0.20 square meters (2.12 square feet)	- μ ~ 20 (over concrete with embedded beams)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/26/84 Bechtel (10)	Stoplog	35 (38 tons)	0.61 m (2 feet)	Tilted drop (45 degrees) on refueling floor	3.5 m/sec (11.4 feet/sec)	Tilted drop; Contact area 0.12 square meters (1.3 square feet)	- Per an unreferenced equation, spalling of concrete will occur at a drop height of approximately 17.8 cm (7 inches) - $\mu \prec$ 1.0, allowable 10 (over concrete, Zones A&B) - Punching shear capacity appears to be high (1.66 MPa [240 psi] from page 12 of calculation) - Calculated punching shear appears to be low (0.70 MPa [101 psi]) - μ = 1.2, allowable 8.72 (over W36, zones A&B) - $\mu \ll$ 1.0 (over concrete, Zone C) - μ ~ 12 , allowable 12 (over W24, Zone C)
Limerick	4/26/84 Bechtel (11)	Stoplog	35 (38 tons)	0.61 m (2 feet)	Flat drop on refueling floor	3.5 m/sec (11.4 feet/sec)	Flat drop; Contact area 1.4 square meters (15 square feet)	- μ = 3.0, allowable 10 (over concrete, Zones A&B) - μ = 1.2, allowable 8.72 (over W36, zones A&B) - Per an unreferenced equation, spalling of concrete will occur at a drop height of approximately 41 cm (16.1 inches) - μ = 1.5, allowable 10 (over concrete, Zone C) - $\mu \prec$ 12, allowable 12 (over W24, zone C)
Limerick	4/26/84 Bechtel (12)	Stoplog	35 (38 tons)	0.46 m (1.5 feet) (in air) 6.9 m (22.5 feet) (water)	Flat drop back into its slot	9.4 m/sec (30.9 feet/sec)	Flat drop; Assume 50 percent contact (0.54 square meters [831.25 square inches])	- Penetration based on impact duration 3.6 cm (1.4 inches) - Penetration based on missiles hitting soils 1.7 cm (0.68 inch)
Limerick	4/26/84 Bechtel (13)	Stoplog	35 (38 tons)	0.56 m (1.83 feet) (in air) 11.6 m (37.75 feet) (water)	Flat drop into the Fuel Pool	10.5 m/sec (34.5 feet/sec)	Flat drop; Assumes 50 percent contact (0.54 square meters [831.25 square inches])	- Penetration based on impact duration 4.3 cm (1.7 inches) - Penetration based on missiles hitting soils 2.2 cm (0.85 inch)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/23/84 Bechtel (14)	Steam dryer assembly	41 (45 tons)	1.8 m (6 feet)	Flat drop on refueling floor	6 m/sec (19.7 feet/sec)	Flat drop; Total contact area 1.9 square meters (3000 square inches); Contact area for slab of interest 0.74 square meters (1140 square inches)	- μ = 3, allowable 10 (over concrete, Zones A&B) - μ = 2.0, allowable 8.72 (over W36, zones A&B) - μ =1.7, allowable 10 (over concrete, Zone C) - μ = 2, allowable 12 (over W24, Zone C)
Limerick	4/26/84 Bechtel (15)	Steam dryer assembly	41 (45 tons)	1.8 m (6 feet)	Tilted drop (17.46 degrees) on refueling floor	6 m/sec (19.7 feet/sec)	Tilted drop; Contact area 0.38 square meters (4.06 square feet)	- $\mu < 1$, allowable 10 (over concrete, Zones A&B) - μ = 9, allowable 8.72 (over W36, Zones A&B)
Limerick	4/26/84 Bechtel (16)	Steam dryer assembly	41 (45 tons)	1.5 m (5 feet)	Tilted drop (14.5 degrees) on refueling floor	5.5 m/sec (17.9 feet/sec)	Tilted drop; Contact area 0.39 square meters (4.18 square feet)	- μ = 8, allowable 8.72 (over W36, Zones A&B) - $\mu < 1$, allowable 10 (over concrete, Zone C) - μ = 50, allowable 12 (over W24, Zone C)
Limerick	4/26/84 Bechtel (17)	Steam dryer assembly	41 (45 tons)	0.9 m (3 feet)	Tilted drop (8.62 degrees) on refueling floor	4.2 m/sec (13.9 feet/sec)	Tilted drop; Contact area 0.68 square meters (7.29 square feet)	- μ = 12, allowable 12 (over W24, Zone C)
Limerick	4/28/84 Bechtel (18)	Steam separator assembly	74 (81.5 tons)	1.5 m (5 feet)	Flat drop on refueling floor	5.5 m/sec (17.9 feet/sec)	Flat drop; Contact area 0.52 square meters (5.61 square feet)	- μ = 2, allowable 10 (over concrete, Zones A&B) - μ = 2.0, allowable 8.72 (over W36, Zones A&B) - μ =1.8, allowable 10 (over concrete, Zone C) - μ = .25, allowable 10 (over W24, Zone C)
Limerick	4/28/84 Bechtel (19)	Steam separator assembly	74 (81.5 tons)	1.5 m (5 feet)	Tilted drop (14.5 degrees) on refueling floor	5.5 m/sec (17.9 feet/sec)	Tilted drop; Contact area 0.46 square meters (4.97 square feet)	- $\mu < 1$, allowable 10 (over concrete, Zones A&B) - μ = 5.5, allowable 20 (two beams, over W36, Zones A&B) - μ =1.5, allowable 10 (over concrete, Zone C - μ = 25, allowable 20 (two beams, over W24, Zone C)
Limerick	4/28/84 Bechtel (20)	Steam separator assembly	74 (81.5 tons)	0.76 m (2.5 feet)	Tilted drop (7.2 degrees) on refueling floor	3.9 m/sec (12.7 feet/sec)	Tilted drop; Contact area 0.52 square meters (5.57 square feet)	- μ = 12, allowable 20 (two beams, over W24, Zone C)
Limerick	4/28/84 Bechtel (21)	Steam separator assembly	74 (81.5 tons)	2.1 m (7 feet)	Flat drop on refueling floor	6.5 m/sec (21.2 feet/sec)	Flat drop; Contact area on slab of interest 0.74 square meters (7.92 square feet)	- μ = 3.5, allowable 10 (over concrete, Zones A&B) - μ = 2.8, allowable 10 (over W36, Zones A&B)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/28/84 Bechtel (22)	Steam separator assembly	74 (81.5 tons)	2.1 m (7 feet)	Tilted drop (20.5 degrees) on refueling floor	6.5 m/sec (21.2 feet/sec)	Tilted drop; Contact area 0.45 square meters (4.88 square feet)	- μ = .7, allowable 10 (over W36, Zone D)
Limerick	5/8/84 Bechtel (23)	Shield plugs	77 (85 tons)	0.9 m (3 feet)	Flat drop on refueling floor	4.2 m/sec (13.9 feet/sec)	Flat drop; Total contact area 38.9 square meters (418.5 square feet); Contact area on slab of interest 16.9 square meters (181.9 square feet)	- $\mu < 10$, allowable 10 (over concrete, Zones A&B)
Limerick	5/8/84 Bechtel (24)	Shield plugs	77 (85 tons)	0.9 m (3 feet)	Tilted drop (5.3 degrees) on refueling floor	4.2 m/sec (13.9 feet/sec)	Tilted drop; Total contact area 0.25 square meters (2.65 square feet)	- $\mu < 1$, allowable 10 (over concrete, Zones A&B) - μ = 4.0, allowable 10 (over W36, Zones A&B) - $\mu < 1$, allowable 10 (over concrete, Zone C - $\mu > 12$, allowable 12 (two beams, over W24, Zone C)
Limerick	5/8/84 Bechtel (25)	Shield plugs	77 (85 tons)	0.61 m (2 feet)	Tilted drop (3.5 degrees) on refueling floor	3.4 m/sec (11.3 feet/sec)	Tilted drop; Total contact area 0.25 square meters (2.67 square feet)	- μ = 12, allowable 12 (two beams, over W24, Zone C)
Limerick	6/17/96 S&L (26)	Shield plug	77 (85 tons)	Various	Tilted blunt drop on drywell head which is 3.8 cm (1.5 inches) thick steel	Various	- Slightly tilted drop - Drywell head material thickness at impact is 3.8 cm (1.5 inches) SA 516 Gr 70 - Postulates the failure of two lifting lugs on the plug - ADINA computer program used to analyze the drywell head under an increasing local load - It is assumed that the plug rotates on a hinge (failure of a lifting lug, not the crane) so only 53 percent of load hits the drywell head - Area of impact 0.49 square meters (754 square inches)	- S&L does not provide an analysis for a sharp (small area) impact - The deflection at maximum strain energy would be approximately 20 cm (8 inches), whereas at the calculated strain energy, the drywell head will deflect approximately 14.7 cm (5.8 inches)
Comanche Peak	12/8/88 SWEC (4)	RCP Assembly	25 (27.6 tons)	Various	203 cm (20 inches) thick RC, slabs S-4 to S-8	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 2.1 m (7 feet) diameter	- Maximum drop height 12.7 cm (5 inches) (Scabbing) - The contact areas were changed in calculation listed as 4-1 below

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Comanche Peak	12/8/88 SWEC (4-1)	RCP Assembly	25 (27.6 tons)	Various	51 cm (20 inches) thick RC, slabs S-4 to S-8	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 0.61 m (2 feet) diameter	- Maximum drop height 35.6 cm (14 inches)
Comanche Peak	12/8/88 SWEC (4-2)	RCP Assembly	25 (27.6 tons)	Various	66 cm (26 inches) thick RC, slabs S-1, 2, 3, and 9	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 0.61 m (2 feet) diameter	- Maximum drop height 99 cm (39 inches)
Comanche Peak	12/8/88 SWEC (5-1)	RCP Stator	21.7 (23.8 tons)	Various	51 cm (20 inches) thick RC, slabs S-4 to S-8	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 1.8 m (6 feet) diameter	- Maximum drop height 12.7 cm (5 inches)
Comanche Peak	12/8/88 SWEC (5-2)	RCP Stator	21.7 (23.8 tons)	Various	66 cm (26 inches) thick RC, slabs S1, 2, 3, and 9	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 1.8 m (6 feet) diameter	- Maximum drop height 38 cm (15 inches
Comanche Peak	12/8/88 SWEC (6-1)	RCP Motor Assembly (Rotor and Stator)	38.5 (42.4 tons)	Various	51 cm (20 inches) thick RC, slabs S-4 to S-8	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 0.61 m (2 feet) diameter	- Maximum drop height 23 cm (9 inches)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Comanche Peak	12/8/88 SWEC (6-2)	RCP Motor Assembly (Rotor and Stator)	38.6 (42.4 tons)	Various	66 cm (26 inches) thick RC, slabs S1, 2, 3, 9, 10	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 0.61 m (2 feet) diameter	- Maximum drop height 0.64 m (2 feet, 1 inch
Comanche Peak	12/8/88 SWEC (7/7A)	RCP Motor Assembly (Rotor and Stator)	25 (27.6 tons)	Various	1.4 m (54 inches) thick RC, slab S-10	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 0.61 m (2 feet) diameter	- Maximum drop height 14.9 m (48 feet, 9 inches) (Scabbing) - Maximum drop height 7.4 m (24 feet, 4 inches) (to reach strain energy maximum)
Comanche Peak	12/8/88 SWEC (8/8A)	RCP Rotor	3 (3.3 tons)	Various	1.4 m (54 inches) thick RC, slab S-10	Various	(General) Strain energy capacity is compared to kinetic energy of the load drop; Assumes a ductility ratio of 10; An elasto-plastic force-deflection diagram is assumed to represent the energy absorbing capacity of the structural component; Missile "area" defined as a 0.61 m (2 feet) diameter	- Maximum drop height 108 m (353 feet, 11 inches) (Scabbing) - Maximum drop height 54 m (176 feet, 8 inches) (to reach strain energy maximum)
Oconee	6/1/82 (1)	Low pressure turbine rotor	126 (138 tons)	9.2 m (30 feet) above turbine deck	Turbine deck floor 29 cm (11.5 inches) thick RC; Second floor 20 cm (8 inches) thick RC; Base floor 1.2 m (48 inches) thick RC	13.4 m/sec (44.0 feet/sec) at impact on turbine deck	Methodology based on Bechtel Power Topical Report, BC-TOP-9 Rev. 2, September 1974 "Design of Structures for Missile Impact"; Rotor falls with it's shaft perpendicular to the floor, flat contact; Ductility ratio of 10	- Perforation depth calculated to be 26 cm (10.31 inches), i.e., the rotor will not go through the turbine deck floor - The drop will result in bending failure of the operating floor slab - The second floor will be penetrated by punching shear - The rotor will penetrate approximate 18 cm (7 inches) into the basement floor - Will not damage any piping greater than 36 cm (14.12 inches) in diameter
Oconee	6/1/82 (2)	Low pressure turbine rotor	126 (138 tons)	23 m (77 feet)	1.5 m (60 inches) thick RC basement floor	21.5 m/sec (70.4 feet/sec)	Methodology based on Bechtel Power Topical Report, BC-TOP-9 Rev. 2, September 1974 "Design of Structures for Missile Impact"; Rotor falls down the equipment hatch	- Penetration depth of rotor 54 cm (21.12 inches) - Some spalling may occur - With not prevent vital embedded systems from performing their safety related functions

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oconee	10/16/75 (3)	Spent fuel cask	22 (24 tons)	14.2 m (46.5 feet) (12 m [40 feet] through water)	Floor of spent fuel pool	16.7 m/sec (54.7 feet/sec)	Allow one trunnion or side of yoke to fail, load stabilizes, and then falls to the SFP floor; Cask hits at approximately 11 degrees; Uses modified Petry formula	- Penetration in steel floor plate 4.9 cm (1.91 inches) (Actual thickness of plate on the floor is 5.7 cm [2.25 inches])
Oconee	5/19/89 (4)	Spent fuel cask	91 (100 tons)	N/A	Floor of spent fuel pool	16.7 m/sec (55.0 feet/sec)	Uses missile impact theory; Very little chance of a large eccentric drop due to gaps between the cask and surrounding equipment; Assumes that the impact is evenly distributed around the cask bottom ring; Assumes that the cask falls through air	- Cask penetration into concrete 29 cm (11.4 inches)
Oconee	5/19/89 (5)	Spent fuel cask	91 (100 tons)	14.2 m (46.5 feet)	Floor of spent fuel pool	14.2 m/sec (46.0 feet/sec)	Uses missile impact theory; Very little chance of a large eccentric drop due to gaps between the cask and surrounding equipment; Assumes that the impact is evenly distributed around the cask bottom ring; Assumes that the cask falls through water; Includes buoyancy and drag effects of water	- Cask penetration into concrete 17 cm (6.8 inches)
Oconee	5/26/89 (6)	Spent fuel cask	N/A	N/A	Floor of spent fuel pool	N/A	Assumes that the largest crack possible would be 0.04 cm (1/64 inch) wide and could include the largest plate in the spent fuel pool (14.4 m [568 inches] in perimeter); Assumes that 12 m (40 feet) of water is in the pool	- The leakage rate was calculated to be 81 liters (21.3 gallons) per day
Oconee	11/21/80 (7)	Spent fuel cask	26.5 (29.1 tons)	8.5 m (27 feet, 9 inches)	Fuel rack	12.9 m/sec (42.3 feet/sec)	Assumes free fall to the rack (no water); Assumes all the kinetic energy is absorbed in part by buoyancy force; Actual crush tests were performed on fuel cans; If cans are damaged, then radioactive gases are released	- 522 cells will be damaged
Oconee	12/2/80 (8)	Spent fuel cask (TN-8)	39.5 (43.4 tons)	8.5 m (27 feet, 9 inches)	Fuel rack	12.9 m/sec (42.3 feet/sec)	Cask hits the side of the spent fuel pool; Assumes all the kinetic energy is absorbed in part by buoyancy force	- 576 cells damaged

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oconee	2/26/88 (9)	Spent fuel cask	N/A	N/A	Fuel rack	N/A	Assumes that a maximum of 1024 assemblies damaged in the units 1 and 2 fuel pool (354 assemblies have less than 1 year decay, the remaining have 1 year decay); Assumes that a maximum of 825 assemblies are damaged in the unit 3 fuel pool (177 assemblies have less than 1 year decay, the remaining have 1 year decay); Assumes that the entire gap activity is released for the effected assembly; No credit is given for HVAC filtration; Beta dose from plume is insignificant	- 0.15 total dose (Rem) for units 1 and 2 - 0.13 total dose (Rem) for unit 3 - 72 thyroid dose (Rem) for units 1 and 2 - 72 thyroid dose (Rem) for unit 3
Diablo Canyon	9/16/86 Bechtel (1)	RCP motor stator	9.1 (10 tons)	30.5 cm (12 inches)	RC slab, infinite thickness	2.5 m/sec (8.0 feet/sec)	Assumes an infinite slab thickness; Assumes missile impact	- Depth of penetration 0.10 cm (0.038 inch)
Diablo Canyon	9/16/86 Bechtel (1)	RCP motor stator	9.1 (10 tons)	30.5 cm (12 inches)	RC slab, 61cm (24 inches thick)	2.5 m/sec (8.0 feet/sec)	Assumes slab thickness of 61 cm (24 inches); Assumes missile impact	- Depth of penetration 0.10 cm (0.038 inch)
Dresden 1	9/28/93 Bechtel (1)	TN-RAM cask	35 (38.5 tons)	61 cm (2 feet) above pool water, 12.5 m (41 feet) of water	Bottom of spent fuel pool, RC 30.5 to 91.4 cm (2 to 3 feet) thick with rock base	13.7 m/sec (44.9 feet/sec)	Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact,” Rev 0; Drops vertically, lands totally flat on cask base	- The concrete base will fail in shear
Dresden 1	9/28/93 Bechtel (2)	TN-RAM cask	35 (38.5 tons)	61 cm (2 feet) above pool water, 12.5 m (41 feet) of water	Bottom of spent fuel pool, RC 30.5 to 91.4 cm (2 to 3 feet) thick with rock base	13.7 m/sec (38.4 feet/sec)	Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact,” Rev 0; Drops horizontally, contact area is calculated assuming a 1.9 cm (0.76 inch) penetration (1.05 square meters [1631 square inches])	- The concrete base will fail in shear

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Dresden 1	9/28/93 Bechtel (3)	TN-9.1 Cask	38 (41.5 tons)	61 cm (2 feet) above pool water, 12.5 m (41 feet) of water	Bottom of spent fuel pool, RC 2-3 feet thick with rock base	11.7 m/sec (38.4 feet/sec)	Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact,” Rev 0; Drops vertically, lands totally flat on cask base	- The concrete base will fail in shear
Dresden 1	9/28/93 Bechtel (4)	TN-9.1 Cask	38 (41.5 tons)	61 cm (2 feet) above pool water, 12.5 m (41 feet) of water	Bottom of spent fuel pool, RC 2-3 feet thick with rock base	8.8 m/sec (29.0 feet/sec)	Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact,” Rev 0; Drops horizontally, contact area is calculated assuming a 1.0 cm (0.406 inch) penetration (1.3 square meters [2044 square inches])	- The concrete base will fail in shear
Dresden 1	10/6/93 Bechtel (5)	TN-9.1 Cask	38 (41.5 tons)	15 cm (6 inches)	20 cm (8 inches) thick RC wall	1.7 m/sec (5.7 feet/sec)	Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact,” Rev 0; ACI 318-83; The cask would have to go over the transfer pool curb which is 25 cm (10 inches), not 15 cm (6 inches) as assumed in the calculation	- Spalling will not occur since wall is >>81 cm (31.9 inches) thick - Speculation is made for drops on the walkway next to the transfer pool
Dresden 1	9/28/93 Bechtel (6)	TN-RAM cask	35 (38.5 tons)	30.5 cm (12 inches)	Washdown area floor, 23 cm (9 inches) thick RC slab	2.4 m/sec (8.0 feet/sec)	Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact,” Rev 0; ACI 318-83	- The concrete base will fail in shear
Dresden 1	10/5/93 Bechtel (7)	TN-RAM	35 (38.5 tons)	See Dresden (3) above	See Dresden (3) above	See Dresden (3) above	See Dresden (3) above; Assumes a redwood crush pad at the bottom of the spent fuel pool; Assumes that the cask lands flat	- Acceptable (59 percent of allowable)
Dresden 1	10/5/93 Bechtel (8)	TN-9.1 Cask	37.8 (41.5 tons)	See Dresden (5) above	See Dresden (5) above	See Dresden (5) above	See Dresden (5) above; Assumes a redwood crush pad at the bottom of the spent fuel pool; Assumes that the cask lands flat	- Acceptable (93 percent of allowable)

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Dresden 1	10/6/94 Vectra (9)	Spent fuel casks	68 - 100 (75 - 110 tons)	3.75 feet in air, 39.25 feet in water	Fuel transfer slab, RC 0.9 m (3 feet) thick	Variable 11.6 to 14.3 m/sec (38 to 47 feet/sec)	ACI-349-85; Bechtel Topical Report, "Design of Structures for Missile Impact, " BC-TOP-9A, Rev 2; Modified Petry formula (missile penetration); Uses a Ballistic Research Lab formula; Assumes a flat cask impact area (100 percent contact) for all equations; Punching shear is the controlling failure mode	- Acceptable for penetration, perforation and spalling (however, impact area of 100 percent was assumed) - Spent fuel pool slab will fail by punching shear - An energy absorbing device would have to be supplied to cover an area of 5.2 m by 3.1 m (17 feet by 10 feet) (Even assuming a flat cask impact area)
Dresden 1	10/6/94 Vectra (10)	Spent fuel cask	100 (110 tons)	N/A	Fuel transfer slab, RC 0.9 m (3 feet) thick	12.2 to 12.8 m/sec (40-42 feet/sec)	Vertical drop; A 45 degree crack will propagate from the outer edge of the cask and completely penetrate the pool floor; Assumes a hole in the pool floor of approximately 14.3 square meters (154 square feet); A coefficient of permeability (0.42 cm/day [0.0137 feet/day]) for a sandy clay soil is assumed	- Maximum leakage calculated to be approximately 10.3 liters (2.7 gallons) per minute which should be easily made up by available water sources
Dresden 2,3	5/21/73 S&L (11)	IF-300 GE cask	63.7 (70 tons)	0.6 m (1.88 feet) in air, 11.5 m (37.75 feet) in water	Spent fuel pool floor, 1.9 m (75 inches) thick RC slab	13.5 m/sec (44.1 feet/sec)	Vertical drop; Modified Petry formula; ACI 318-71; Assumes a flat cask impact (100 percent impact area of the fins, 0.29 square meters [445.5 square inches])	- Penetration in slab 25.5 cm (10.03 inches) - Load factor of 2 against punching shear - Load factor of 1.44 against cracking
Dresden 2,3	5/21/73 S&L (12)	IF-300 GE cask	63.7 (70 tons)	0.6 m (1.88 feet) in air, 11.5 m (37.75 feet) in water	Spent fuel pool floor, 1.9 m (75 inches) thick RC slab	13.4 m/sec (43.9 feet/sec)	Horizontal drop; Modified Petry formula; ACI 318-71; Assumes a reduce contact area of 0.65 square meters (1008 square inches)	- Penetration in slab 11.4 cm (4.5 inches) - Load factor of 1.5 - Load factor of 2 against punching shear
Dresden 2,3	5/21/73 S&L (13)	IF-300 GE cask	63.7 (70 tons)	N/A	Decontamination pit	N/A	Vertical drop; ACI 318-71; Due to the complex shape, the slab was transformed into an equivalent fixed ended beam of 2.9 m (9.5 feet) in width	- The maximum drop height was calculated to be 28.3 cm (11.15 inches) - It was recommended that the cask be raised a maximum of 23 cm (9 inches) for safe cleaning operation, and 15 cm (6 inches) while traveling to and from the decontamination pit

Table F1: Heavy load drop calculations (continued)

PLANT	CALC DATE	LOAD	WT (metric tons)	DROP HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Dresden 2,3	5/21/73 S&L (14)	IF-300 GE cask	63.7 (70 tons)	N/A	Travel path between the decontamination pit and the spent fuel pool over the torus	N/A	An extension from (13) above; Vertical drop	- Two pathways were analyzed (slabs and beams, and over beams) - The pathway over beams was the most desirable, which indicated that the cask could be raised to a maximum height of 56 cm (22 inches) - A conservative lift height of 15 cm (6 inches) was made
Dresden 2,3	7/2/81 S&L (15)	N/A	N/A	29 m (95.5 feet)	Drop down the reactor building equipment hatch to the main floor over the torus	23.9 m/sec (78.4 feet/sec)	Assume the dropped load has a diameter of 45.7 cm (18 inches); RC slab, 61 cm (24 inches) thick; Assumes concrete will fail at approximately 5.9 E05 kgs (1300 kips), then calculates the penetration depth into the concrete from an initial height of 29 m (95.5 feet)	- To prevent scabbing of a 61 cm (24 inches) thick floor, the missile penetration depth cannot be > 8.3 cm (3.27 inches) - Maximum load drop (from 29 m [95.5 feet]) with no scabbing of a 61 cm (24 inches) thick slab calculated to be 0.9 metric ton (1 ton)
Dresden 2,3	7/2/81 S&L (16)	N/A	N/A	29 m (95.5 feet)	Drop down the reactor building equipment hatch to the main floor over the torus	23.9 m/sec (78.4 feet/sec)	Assume the dropped load has a diameter of 45.7 cm (18 inches); RC slab, 81.3 cm (24 inches) thick; Assumes concrete will fail at approximately 5.9 E05 kgs (1300 kips), then calculates the penetration depth into the concrete from an initial height of 29 m (95.5 feet)	- To prevent scabbing of a 61 cm (24 inches) thick floor, the missile penetration depth cannot be > 8.3 cm (3.27 inches) - Maximum load drop with no perforation of a 61 cm (24 inches) thick slab is calculated to be 5.2 metric tons (5.75 tons)
Dresden 2,3	7/2/81 S&L (17)	N/A	N/A	29 m (95.5 feet)	Drop down the reactor building equipment hatch to the main floor over the torus	23.9 m/sec (78.4 feet/sec)	Assume the dropped load has a diameter of 45.7 cm (18 inches); RC slab, 81.3 cm (32 inches) thick; Assumes concrete will fail at approximately 5.9 E05 kgs (1300 kips), then calculates the penetration depth into the concrete from an initial height of 29 m (95.5 feet)	- To prevent scabbing of a 81 cm (32 inches) thick floor, the missile penetration depth cannot be > 8.3 cm (3.27 inches) - Maximum load drop with no scabbing of a 81 cm (32 inches) thick slab is calculated to be 1.8 metric tons (2 tons)
Dresden 2,3	7/2/81 S&L (18)	N/A	N/A	29 m (95.5 feet)	Drop down the reactor building equipment hatch to the main floor, over the torus	23.9 m/sec (78.4 feet/sec)	Assumes the dropped load has a diameter of 24 cm (24 inches); RC slab, 81 cm (32 inches) thick; Assumes concrete will fail at approximately 5.9 E05 kgs (1300 kips), then calculates the penetration depth into the concrete from an initial height of 29 m (95.5 feet)	- To prevent scabbing of a 81 cm (32 inches) thick floor, the missile penetration depth cannot be > 11.7 cm (4.62 inches) - To produce a penetration depth of 11.7 cm (4.62 inches) was calculated to be 2.4 E03 kgs (5.36 kips)

Table F2: Heavy load calculation abbreviations

ACI	American Concrete Institute	psi	pounds per square inch
B	bottom	RC	reinforced concrete
		RCP	reactor coolant pump
cm	centimeter	RPV	reactor pressure vessel
E/W	east west	S&L	Sargent and Lundy
EQE	engineering company	sec	second
		SWEC	Stone and Webster Engineering Company
ft	feet		
GE	General Electric	T	top
		TN	vendor name of cask
HVAC	heating, ventilation, and air conditioning	TVA	Tennessee Valley Authority
		Vectra	engineering company
k	1000		
kg	kilogram	W	structural beam web
kip	1000 pounds		
		μ	ductility ratio
m	meter		
MPa	megapascal		
N/S	north south		
N/A	not applicable		
NAM	vendor name of cask		

Appendix G

NRC Generic Communications Involving Crane Operating Experience

INTRODUCTION

This appendix lists NRC generic communications that have been issued since 1976 that involved crane activities. Table G1: “NRC Generic Communications Involving Crane Operation Issues,” lists 29 NRC generic communications (including their supplements) involving load movement issues dating from 1976. Several generic communications involved crane weaknesses, but most involved crane operations or crane program implementation weaknesses which were the focus of Phase I of NUREG-0612 issued in 1980. Phase I consists of the seven good practices listed in Section 5.5.1 of NUREG-0612. As stated in Section 5.5.1, “... 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.” Phase I criteria include the areas of training, safe load paths, load handling procedures, requirements for special lifting devices, crane design, inspection and maintenance.

Table G1: NRC Generic Communications Involving Crane Operation Issues

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
CR 76-01	7/27/76	Crane Hoist Control - Circuit Modifications	The hoist control system at Dresden Units 2 and 3, and Quad Cities Units 1 and 2 was marginal. On several occasions when the low speed motor was stopped in the lowering mode, the solenoid circuit contacts arced resulting in power being supplied to the solenoids long enough so that the load dropped some distance before the brakes engaged. Over travel of as much as 15 inches was reported, but no damage to hoist or load was found.	1) Determine and report to this office within 90 days the following information: (a) Have you made, or do you plan to make modifications to the hoist control for your installed cranes similar to the described modifications? (b) If such modifications have been made, or are planned, identify changes required in brake power and control circuitry. (c) What steps have been taken or are planned, to provide assurance that brake power contactors are adequate for the service? 2) If modifications are planned, provide the schedule for completion and a brief description of your plans for design review and functional testing.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
BL 76-07	7/27/76	Crane Hoist Control - Circuit Modifications	The hoist control system at Dresden Units 2 and 3, and Quad Cities Units 1 and 2 was marginal. On several occasions when the low speed motor was stopped in the lowering mode, the solenoid circuit contacts arced resulting in power being supplied to the solenoids long enough so that the load dropped some distance before the brakes engaged. Over travel of as much as 15 inches was reported, but no damage to hoist or load was found.	1) Determine and report to this office within 20 days the following information: (a) Have you made, or do you plan to make modifications to the hoist control for your installed cranes similar to the described modifications? (b) If such modifications have been made, or are planned, identify changes required in brake power and control circuitry. (c) What steps have been taken or are planned, to provide assurance that brake power contactors are adequate for the service? 2) If modifications are planned, provide the schedule for completion and a brief description of your plans for design review and functional testing.
CR 77-12	9/15/77	Dropped Fuel Assemblies at BWR Facilities	Several events are described involving dropped fuel assemblies at Pilgrim, Millstone Unit 1, Humbolt Bay, Duane Arnold, Brunswick Unit 2, and Peach Bottom Unit 3.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
GL 78-16	5/16/78	Request for Information On Control of Heavy Loads Near Spent Fuel Pools	[For licensees in the Systematic Evaluation Program]	(1) Provide a diagram which illustrates the physical relation between the reactor core, the fuel transfer canal, the spent fuel storage pool and the set down, receiving or storage areas for any heavy loads moved on the refueling floor. (2) Provide a list of all objects that are required to be moved over the reactor core (during refueling), or the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram of the movement path utilized (including carrying height) and the frequency of movement. (3) What are the dimensions and weights of the spent fuel casks that are or will be used at your facility? (4) Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC staff. (5) Identify any heavy loads that are carried over equipment required for the safe shutdown of a plant that is operating at the time the load is moved. Identify what equipment could be affected in the event of a heavy load handling accident (piping, cabling, pumps, etc.) And discuss the feasibility of such an accident affecting this equipment. Describe the basis for your conclusions. (6) If heavy loads are required to be carried over the spent fuel storage pool or fuel transfer canal at your facility, discuss the feasibility of a handling accident which could result in water leakage severe enough to uncover the spent fuel. Describe the basis for your conclusions. (7) Describe any design features of your facility which affect the potential for a heavy load handling accident involving spent fuel, e.g., utilization of a single failure-proof crane. (8) Provide copies of all procedures currently in effect at your facility for the movement of heavy loads over reactor core during refueling, the spent fuel storage pool or equipment required for the safe shut-down of a plant that is operating at the time the move occurs. (9) Discuss the degree to which your facility complies with the eight (8) regulatory positions delineated in Regulatory Guide 1.13 (Rev. 1, December 1975) regarding Spent Fuel Storage Facility Design Basis.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
GL 78-15	5/17/78	Request for Information On Control of Heavy Loads Near Spent Fuel	[For licensees except those in the Systematic Evaluation Program]	(1) Provide a diagram which illustrates the physical relation between the reactor core, the fuel transfer canal, the spent fuel storage pool and the set down, receiving or storage areas for any heavy loads moved on the refueling floor. (2) Provide a list of all objects that are required to be moved over the reactor core (during refueling), or the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram of the movement path utilized (including carrying height) and the frequency of movement. (3) What are the dimensions and weights of the spent fuel casks that are or will be used at your facility? (4) Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC staff. (5) Identify any heavy loads that are carried over equipment required for the safe shutdown of a plant that is operating at the time the load is moved. Identify what equipment could be affected in the event of a heavy load handling accident (piping, cabling, pumps, etc.) and discuss the feasibility of such an accident affecting this equipment. Describe the basis for your conclusions. (6) If heavy loads are required to be carried over the spent fuel storage pool or fuel transfer canal at your facility, discuss the feasibility of a handling accident which could result in water leakage severe enough to uncover the spent fuel. Describe the basis for your conclusions. (7) Describe any design features of your facility which affect the potential for a heavy load handling accident involving spent fuel, e.g., utilization of a single failure-proof crane. (8) Provide copies of all procedures currently in effect at your facility for the movement of heavy loads over reactor core during refueling, the spent fuel storage pool or equipment required for the safe shut-down of a plant that is operating at the time the move occurs. (9) Discuss the degree to which your facility complies with the eight (8) regulatory positions delineated in Regulatory Guide 1.13 (Rev. 1, December 1975) regarding Spent Fuel Storage Facility Design Basis.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
GL 78-17	6/12/78	Corrected Letter On Heavy Loads Over Spent Fuel	[For licensees except those in the Systematic Evaluation Program]	(1) Provide a diagram which illustrates the physical relation between the reactor core, the fuel transfer canal, the spent fuel storage pool and the set down, receiving or storage areas for any heavy loads moved on the refueling floor. (2) Provide a list of all objects that are required to be moved over the reactor core (during refueling), or the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram of the movement path utilized (including carrying height) and the frequency of movement. (3) What are the dimensions and weights of the spent fuel casks that are or will be used at your facility? (4) Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC staff. (5) Identify any heavy loads that are carried over equipment required for the safe shutdown of a plant that is operating at the time the load is moved. Identify what equipment could be affected in the event of a heavy load handling accident (piping, cabling, pumps, etc.) and discuss the feasibility of such an accident affecting this equipment. Describe the basis for your conclusions. (6) If heavy loads are required to be carried over the spent fuel storage pool or fuel transfer canal at your facility, discuss the feasibility of a handling accident which could result in water leakage severe enough to uncover the spent fuel. Describe the basis for your conclusions. (7) Describe any design features of your facility which affect the potential for a heavy load handling accident involving spent fuel, e.g., utilization of a single failure-proof crane. (8) Provide copies of all procedures currently in effect at your facility for the movement of heavy loads over reactor core during refueling, the spent fuel storage pool or equipment required for the safe shut-down of a plant that is operating at the time the move occurs. (9) Discuss the degree to which your facility complies with the eight (8) regulatory positions delineated in Regulatory Guide 1.13 (Rev. 1, December 1975) regarding Spent Fuel Storage Facility Design Basis.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 80-01	1/4/80	Fuel Handling Events	Two recent fuel handling events are described where there were no interlocks to control or limit the movement of nuclear fuel in the reactor building.	None.
CR 80-13	5/28/80	Grid Strap Damage in Westinghouse Fuel Assemblies	Describes damage to 31 assemblies at Salem Unit 1 which appeared to be the result of corner to corner interaction of the grid straps of diagonally adjacent fuel assemblies during vertical loading and unloading assembly movements.	None.
GL 80-113	12/22/80	Control of Heavy Loads	Discusses the closure of Unresolved Safety Issue A-36, “Control of Heavy Loads Near Spent Fuel” through the issuance of NUREG-0612.	Requested licensees to (1) submit a report documenting required changes and modifications, and how the guidelines of NUREG-0612 will be satisfied, (2) furnish confirmation within six months that implementation of those changes and modifications you find are necessary will commence as soon as possible without waiting for staff review, so that all such changes, beyond the above interim actions, will be completed within two years of submittal, (3) Furnish justification within six months for any changes or modifications that would be required to fully satisfy the NUREG-0612 guidelines you believe are not necessary.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
GL-81-07	2/3/81	Control of Heavy Loads	Discusses missing information that was referred to but left out of GL 80-113. Clarifies the information that should be provided to address heavy loads and postulated load drops.	(1) 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. (2) 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. (3) 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 consideration of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects. (4) Provide an evaluation comparing the results of the analysis to Criterion I of NUREG-0612. If the postulated heavy-load-drop accident analyzed bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 81-23	8/4/81	Fuel Assembly Damaged Due to Improper Positioning of Handling Equipment	Provides two examples of fuel handling deficiencies, one at Cook Unit 1, and another at Point Beach Unit 2. At the Cook facility, an assembly was damaged when it struck the ledge on the refueling cavity floor just outside the reactor vessel area. Several rods in the fuel assembly were damaged, and one rod was dislodged and fell from the assembly onto the refueling cavity floor. No radiation release occurred as a result of the damaged rods.	None.
IN 83-35	5/31/83	Fuel Movement With Control Rods Withdrawn at BWRs	Provides examples at Brunswick and Duane Arnold of situations where fuel movements were made while control rods were not fully inserted.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 83-71	10/27/83	Defects in Load-Bearing Welds on Lifting Devices for Vessel Head and Internals.	A lifting device for the reactor vessel head and internals supplied to Florida Power Corporation by Babcock and Wilcox was found to have weld defects in load-bearing welds. Lifting devices must meet the requirements of ANSI N14.6.	None.
GL 83-42	12/19/83	Clarification to Generic Letter 81-07 Regarding Response to NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.”	In the course of reviewing crane designs against NUREG-0554, “Single Failure Proof Cranes,” the NRC identified concerns of a generic nature (i.e., inability of the crane to support a load or the failure of the brake to set) which indicate that NUREG-0554 until revised, may be deficient in assuring single failure proof cranes.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
GL 85-11	6/28/85	Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612	Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase II), further action is not required to reduce the risks associated with the handling of heavy loads. Therefore, a detailed Phase II review of heavy loads is not necessary and Phase II is considered completed. A cost benefit analysis for upgrading polar cranes to single failure proof indicated that the NRC can not perceive a significant enough benefit in the conversion to a single failure proof polar crane to warrant the high costs estimated at $30 million.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 85-12	2/11/85	Recent Fuel Handling Events	Lists several fuel handling events ranging from fuel assembly drops, to collisions, to stuck elements. The events were noted at Hatch Unit 1, Millstone Unit 2, Monticello, Palisades, Turkey Point Unit 4, and Cook Unit 1.	None.
IN 86-06	2/3/86	Failure of Lifting Rig Attachment While Lifting the Upper Guide Structure at St. Lucie Unit 1	A rigging failure at St. Lucie Unit 1 is described. While performing a lift of the upper guide structure weighing approximately 50 tons, a rigging device joint (secured by a bolt) failed because it was improperly made up (e.g., lacked adequate thread engagement). The upper guide structure tilted approximately 6 inches when the bolt failed. No damage was done.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 86-58	7/11/86	Dropped Fuel Assembly	The IN describes a dropped fuel assembly event at Haddam Neck. During the lift of the upper core support structure weighing approximately 57,000 pounds, a fuel assembly stuck to the structure because of a bent fuel assembly locating pin. The assembly fell off when the load was moved laterally. The dropped assembly and the two fuel bundles that it impacted were damaged, but there was no radiation release.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 90-77	12/12/90	Inadvertent Removal of Fuel Assemblies from the Reactor Core	Provides examples (Indian Point Unit 3, Palisades, and Byron Unit 2) where fuel assemblies were stuck to the upper core support structures when the upper support structures were removed during a lift, or where guide pins were damaged during the lift.	None.
IN 90-77-S1	2/4/91	Inadvertent Removal of Fuel Assemblies from the Reactor Core	Provides an additional information on the event at Indian Point Unit 3 on 10/4/90 that was developed by an AIT.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 92-13	2/18/92	Inadequate Control Over Vehicular Traffic at Nuclear Power Plant Sites	This IN describes four events where a loss or partial loss of offsite power occurred because of the contact of vehicles with electrical equipment. Diablo Canyon 1 (caused by a mobile crane), Palo Verde 3 (caused by a mobile crane), and Fermi 2 (caused by a mobile crane), and Vogtle (caused by a truck)	None.
IN 94-13	2/22/94	Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment	Provides several examples of fuel movement problems at Vermont Yankee, Peach Bottom Unit 3, Susquehanna Unit 1, Susquehanna Unit 2, and Nine Mile Point Unit 2.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 94-13-S1	6/28/94	Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment	Provides an additional example of a fuel movement issue at Waterford. An unknown object was found attached to the fuel handling machine.	None.
IN 94-13-S2	11/28/94	Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment	This supplement discusses events that occurred at the Hatch facility that demonstrate the potential for equipment damage and personnel hazards as a result of inadequately supervised contractor activities on the refueling floor.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
Bulletin 96-02	4/11/96	Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment	This Bulletin was initiated in response to issues surrounding issues involving heavy load movements at Oyster Creek. The Bulletin alerts licensees to the importance of complying with existing regulatory guidelines associated with the control and handling of heavy loads at nuclear power plants while the plant is operating, and to remind licensees of their responsibilities for ensuring that heavy load activities carried out under their license are performed safely and within the requirements specified under Title 10 of the CFR.	Licensees were requested to review plans and capabilities for handling heavy loads while the reactor is at power in accordance with existing regulatory guidelines. Determine whether the activities are within the licensing basis and, if necessary, submit a license amendment request. Determine whether changes to Technical Specifications will be required in order to allow the handling of heavy loads over fuel assemblies in the spent fuel pool.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 96-26	4/30/96	Recent Problems With Overhead Cranes	This IN describes two events, one involving a crane rail original design problem concerning polar crane rails at Trojan, and a second at Prairie Island involving a mis-calibrated overload switch that was discovered during a cask lift.	None.
IN 97-51	7/11/97	Problems Experienced With Loading and Unloading Spent Nuclear Fuel Storage and Transportation Casks	Makes reference to several heavy loads issues, including Bulletin 96-02. The IN describes some of the problems encountered by licensees in preparing for or actually performing the loading or unloading of storage or transportation casks.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 99-15	5/27/99	Misapplication of 10 CFR Part 71 Transportation Shipping Cask Licensing Basis to 10 CFR Part 50 Design Basis	This IN was issued to ensure that where 10 CFR Part 71 licensing-basis information is being relied upon to satisfy the design basis for 10 CFR Part 50, licensee should ensure that the Part 71 information is adequately supported to satisfy the requirements of Part 50. Information provided by a vendor should be consistent by may not always be site specific. Where a site-specific analysis is needed to support FSAR statements, vendor information should be supplemented with the necessary additional analysis.	None.

Table G1: NRC Generic Communications Involving Crane Operation Issues (Continued)

GENERIC COM	ISSUE DATE	TITLE	ABSTRACT	ACTION REQUESTED
IN 2002-09	2/13/02	Potential for Top Nozzle Separation and Dropping of a Certain Type of Westinghouse Fuel Assembly	This IN indicates that the type of fuel assembly that may have separation problems was last manufactured almost 20 years ago. However, these fuel assemblies may be involved in cask loading operations. An event that occurred at North Anna is provided.	None.