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NUREG-XXXX

Technical Assessment
Generic Issue 186:
Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants

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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 1999. Crane operating experience information came 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. In addition, major crane operating experience reports issued by the New Mexico Environmental Evaluation Group, the Department of Energy, the Department of the Navy, the California Division of Occupational Safety and Health, and Appendix A to NUREG-1738 titled *Structural Integrity of Spent Fuel Pool Structures Subject to Heavy Loads Drops*, have been included as appendices to this report to provide additional insights.

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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 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. 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. 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 a Technical Assessment of the GI, and including a comprehensive retrieval and analysis of information on crane operating experience in the U.S. nuclear industry, documented in the NRC's NUDOCS system, from 1968 through October 1999. Actual crane operating experience was obtained from eight nuclear facilities representing approximately 14 percent of the available operating experience in the U.S.

The study found that there were no risk significant events involving loads of approximately 30 tons or greater occurred at any U.S. nuclear plant having an operating license. There have been injuries and deaths caused by crane operation, but no radiation releases or risk to the health and safety of the public. There were six potentially risk-significant crane events involving a loss or partial loss of offsite power caused by mobile cranes. Two of the six events (Palo Verde and Diablo Canyon) resulted in Augmented Inspection Team (AIT) inspections, however, none of the six crane events met the minimum risk threshold requirements to be classified as an Accident Sequence Precursor (ASP) event. A review of all ASP data for the period 1985 through 1999 indicated that there were no crane events that were classified as ASP events (e.g. a minimum conditional core damage probability of $1x10^{-6}$ or greater).

There were several indicators that crane operating performance has greatly improved since in issuance of NUREG-0612 in 1980. While the number of operating plants has almost doubled since 1980, the number of load drops and load slips has remained somewhat constant. The number of deaths and injuries dramatically decreased during post 1980 when compared to pre-1980 rates, especially when considering the rapid increase in the number of licensed operating plants. Generic Letter 85-11 also indicated 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... Therefore, a detailed Phase II review (o)f heavy loads is not necessary and Phase II is considered completed."

The study showed that there were inconsistent licensee approaches to load drop calculation methodologies, assumptions, and load lift height restrictions. Reviews of load drop calculations obtained from each facility that was visited indicated that calculational methodologies and assumptions varied greatly from licensee to licensee, producing radically different end results. Heights of load drops, plant locations for postulated load drops, contact area at impact, materials property values, and weights of loads varied greatly. The Oyster Creek calculation for a drop of a 45 ton fuel cask over a reinforced concrete slab, 16 inches thick, was the most restrictive, with an allowable drop height of 2.77 inches. Some facilities performed load drop calculations using equations that were intended for ballistic type situations meant for high velocity and low mass. In addition, according to licensee responses to NRC Bulletin 96-02, only 8 licensees indicated that a consequence analysis had been done at their facility for heavy load drops.

There was some confusion regarding single-failure-proof crane classification. 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.

The study found that NRC generic communication documents concerning heavy load drop issues have not been fully effective. Despite existing NRC regulatory requirements and reminders through the generic communication process, fundamental questions still remain: (1) What is the acceptable load lift height for various loads, (2) What are the necessary crane program requirements, and (3) What are the requirements for load movements at power vs. shutdown. Several regulatory documents have been issued that relate to very heavy loads: Unresolved Safety Issue (USI) A-36, *Control of Heavy Loads near Spent Fuel* including followup documents NUREG-0554, *Single-Failure-Proof Cranes for Nuclear Power Plants,* and NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants;* Generic Letter 80-113 (originally unnumbered), *Control of Heavy Loads;* Generic Letter 81-07, Control of Heavy Loads; Generic Letter 85-11, *Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612;* and Bulletin 96-02, *Movement of Heavy Loads Over Spent Fuel in the Reactor Core, or Over Safety-related Equipment.* With the exception of GL-85-11, the primary message in each of these documents was to request licensees to assess their heavy loads programs and make whatever changes that were found to be necessary. In addition to the major heavy load documents listed above, other generic communication was issued by the NRC in addressing heavy load issues such as: IN 80-01, *Fuel Handling Events*; IN 81-23, *Fuel Assembly Damaged Due to Improper Positioning of Handling Equipment*; IN 85-12, Recent Fuel Handling Events; IN 86-06, Failure of Lifting Rig Attachment While Lifting the Upper Guide Structure At St. Lucie Unit 1; IN 86-58, *Dropped Fuel Assembly*; IN 92-13, *Inadequate Control Over Vehicular Traffic at Nuclear Power Plant Sites*; IN 96-26, *Recent Problems with Overhead*

Cranes; and IN 97-51, *Problems Experienced with Loading and Unloading Spent Nuclear Fuel Storage and Transportation Casks*.

ACKNOWLEDGMENTS

This report benefitted greatly through the willingness of several staff at Brown's Ferry, Comanche Peak, Diablo Canyon, Dresden, Grand Gulf, Limerick, Oconee, Oyster Creek, and Palo Verde nuclear power plants, 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
AE	architect engineer
AIT	Augmented Inspection Team
ASP	accident sequence precursor
BWR	boiling water reactor
CCDP	conditional core damage probability
CFR	Code of Federal Regulations
CRDM	control rod drive mechanism
DOE	U.S. Department of Energy
EDG	emergency diesel generator
EEG	Environmental Evaluation Group
EQE	EQE International
FSAR	final safety analysis report
GI	generic issue
GL	generic letter
HT	height
HVAC	heating, ventilation, and air conditioning
IC	Interaction Coefficient
IMIS	integrated management information system
IN	information notice
ISFSI	independent spent fuel storage installation
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
PWR	pressurized water reactor
RC	reinforced concrete
RCP	reactor coolant pump
RES	Office of Nuclear Regulatory Research (NRC)
RHR	residual heat removal

ABBREVIATIONS (Continued)

RPV	reactor pressure vessel
RWCU	reactor water clean up
SFP	spent fuel pool
SSE	safe shutdown equipment
SWEC	Stone and Webster Engineering Company
UFSAR	Updated Final Safety Analysis Report
USI	unresolved safety issue
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 in 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 80-113 (originally unnumbered), *Control of Heavy Loads*, (December 1980)

  Generic Letter (GL) 80-113 requested that licensees review their controls for handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 are present at their facilities, and to identify the changes and modifications that would be required in order to fully satisfy these guidelines.

- Generic Letter 81-07, Control of Heavy Loads (February 3, 1981)

  GL-81-07 clarifies parts of GL-80-113 and requests that additional information be provided for analyses. Licensees were requested to provide additional items such as initial conditions/assumptions of postulated load drops, methods used in the analysis, an analysis that demonstrates that ceilings are not penetrated, and an analysis to demonstrate that post-accident dose will be well within 10 CFR Part 100 limits.

- 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 was initiated because of load drop analysis performed by the Oyster Creek nuclear power plant. The bulletin: (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.

  This bulletin required licensees to:

  - report within 30 days of the date of the bulletin, indicating the review of their plans and capabilities to handle heavy loads while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) in accordance with existing regulations, and

  - provide a statement of the capability of performing the actions necessary for safe shutdown in the presence of radiological source term that may result from a breach of the dry storage cask, damage to the fuel, and damage to safety-related equipment as a result of a load drop inside the facility.

  Responses to Bulletin 96-02 (See section 5.0) revealed that although some plants may have reduced the potential for load drops through upgrades to the lifting system, a majority of the plants either did not evaluate or were uncertain of their plans to evaluate the potential consequences for heavy load drops.

  The staff closed its review of the responses to the bulletin generically and committed to perform more detail reviews of licensees' load handling operations on a plant specific basis.

- 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 entire Nuclear Documents System (NUDOCS) database was searched for documents relating to cranes for the period 1968 through 1999. 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 Crane Event Database Categories and Subcategories

To analyze crane issues recorded in NUDOCS, several general categories were established, most with several subcategories. Once this information was input, sorts were performed to look for trends and patterns.

Table 1: Crane event database categories and subcategories

General Event Category	Event Subcategories
Plant and event date	Docket, plant name, event year, event month
Crane type	Reactor building, polar, auxiliary, refueling/manipulator, spent fuel pool, tower, mobile, other
Crane component deficiency	Structure, control, brakes, rails, fasteners, unknown, none
Reported administrative cause for 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, loss or partial loss of power, none
Load description for slip or drop events	Load description (component and weight), height of drop or slip

2.2 Analysis of Crane Events Documented in NUDOCS

A review of crane documents in NUDOCS for the period 1968 through 1999 resulted in 294 different issues. Depending on the severity of each issue, each issue may be discussed in several documents. Most are administrative (not following a procedure, load path issues, noncompliance with technical specifications, inadequate crane operational testing prior to use, etc.) and few relate to problems encountered when lifting loads of approximately 30 tons or more. The following figures not only include a wide span of operating experience, but also include a wide variety of crane types, some of which are not used at operating nuclear facilities today. Figures 1 through 11 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 discusses a subset of information contained in this section in that it contains an analysis of crane operating experience at nuclear power plants that have an operating license, and only for those loads that are classified in this report as “very heavy” (greater than approximately 30 tons).

2.2.1 Reported Crane Issues

Figure 1, *Crane issues documented in NUDOCS (1968-1999)*, shows the total number of reported crane issues in two year increments. Crane issues were reported by individual licensees, through NRC documents and inspection reports, by vendors, and the public. It is unknown how many crane events or issues did not get reported and were not entered in NUDOCS. Figure 1 also shows the total number of nuclear power plants that were licensed to operate during each of the two-year time periods. A statistical correlation exists between the number of plants that have an operating licensee and the number of reported crane issues (correlation coefficient of .77). Overall, this correlation would tend to indicate that the number of crane issues is proportional to the number of licensees with an operating license. However, the severity of crane issues as measured by the number of drops, slips, deaths or injuries has not been proportional to the increase in the number of operating nuclear power plants (see figures 8 and 10).

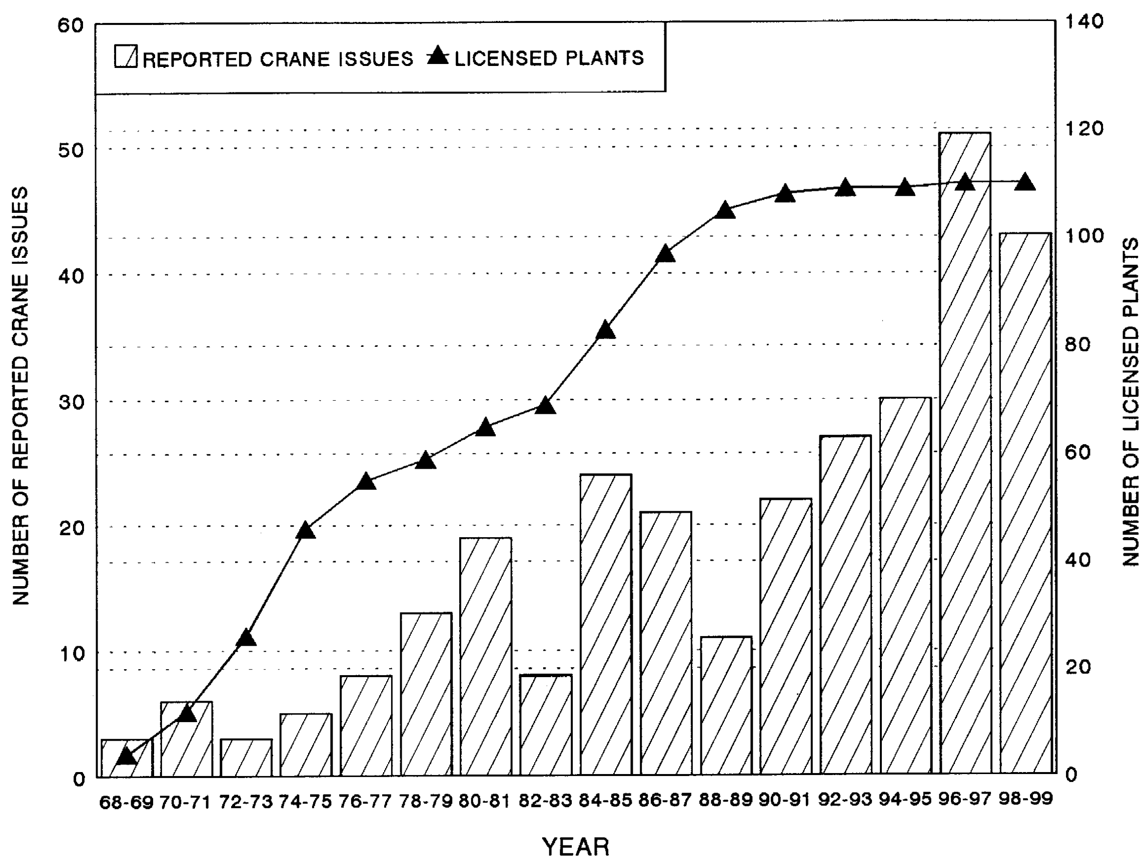

Figure 1: Crane issues documented in NUDOCS (1968-1999)

2.2.2 Crane Reports Due to Not Following Procedures

Figure 2, *Crane issues due to not following procedures (1968-1999)*, shows the percentage of crane issue reports that were caused by not following procedures. As shown in the figure, the percentage of crane issue reports caused by not following procedures has been cyclic, with an overall average of approximately 37 percent.

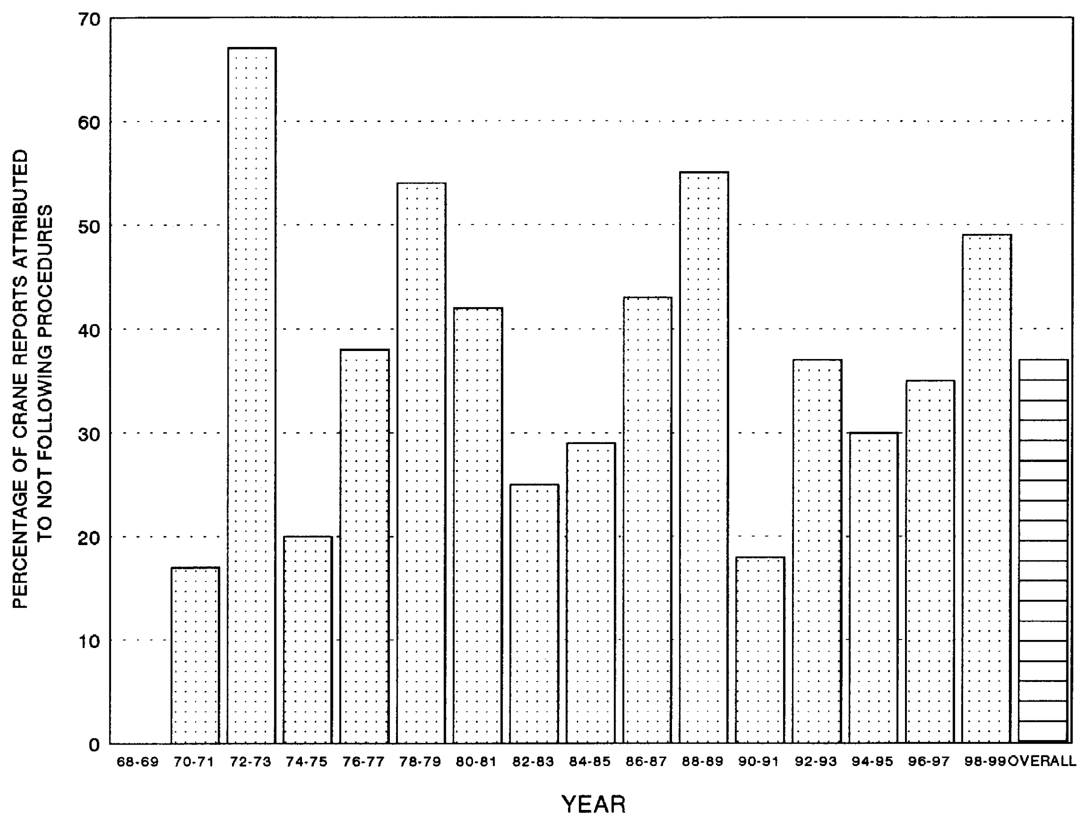

Figure 2: Crane issues due to not following procedures (1968-1999)

2.2.3 Crane Event Distribution by Crane Type

For the 294 reported crane issues during the period 1968 through 1999, Figure 3, *Crane issue distribution by crane type (1968-1999)*, shows the distribution by 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, refueling/manipulator (RF/MC), reactor (RX) building, 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.

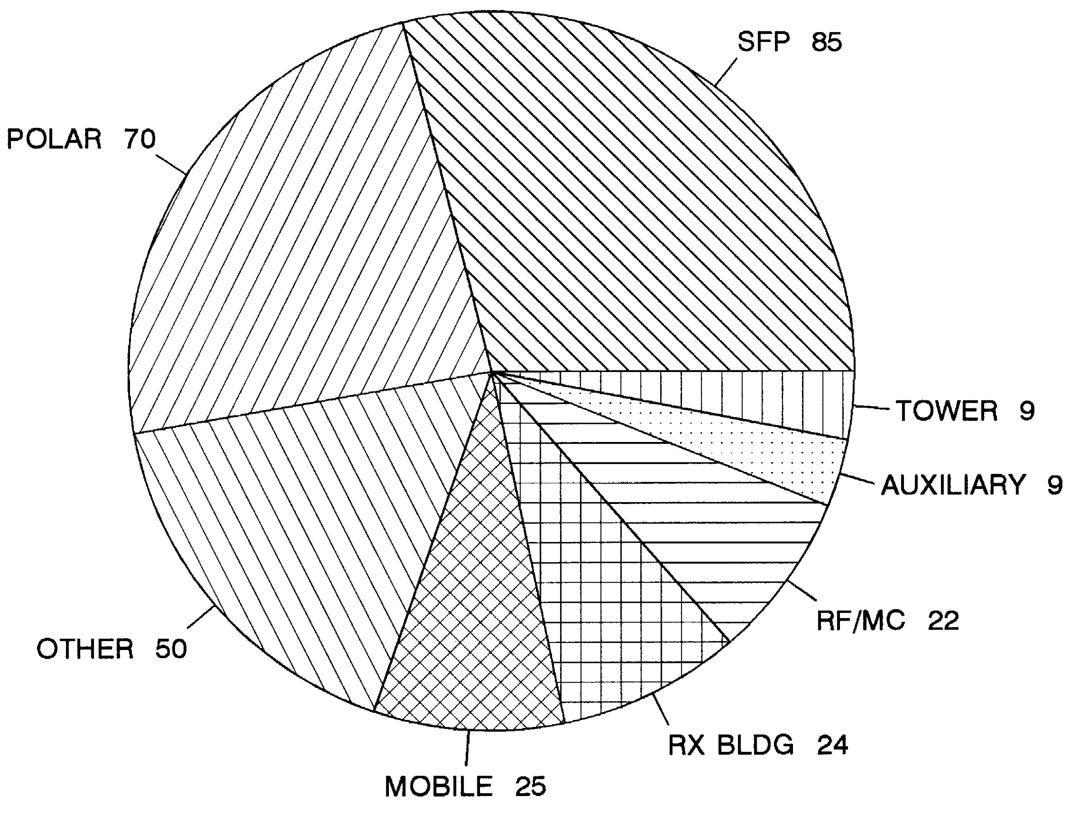

Figure 3: Crane issue distribution by crane type

2.2.4 Crane Events Due to Hardware Deficiencies

As shown in Figure 4, *Crane events due to hardware deficiencies (1968-1999)*, of the 294 crane issues, 112 involved actual 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. As can be seen from the figure, 179 of 294 reported crane issues did not involve a hardware deficiency.

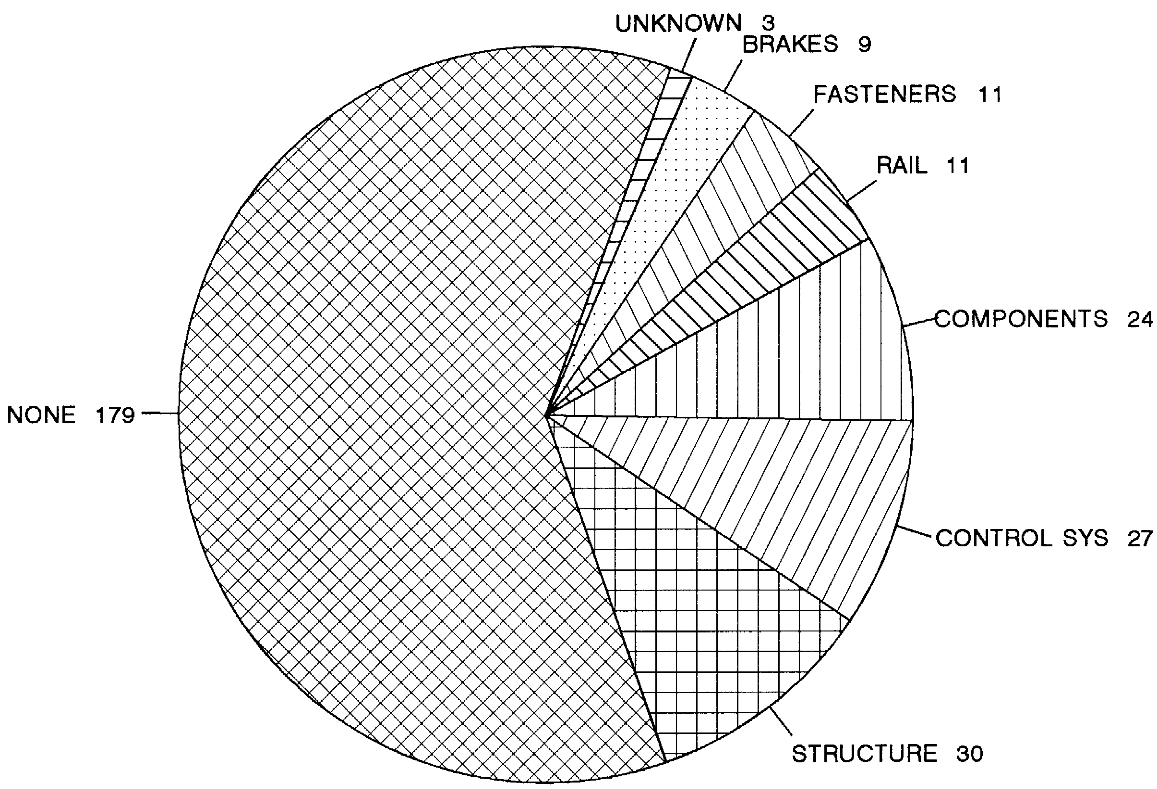

Figure 4: Crane events due to hardware deficiencies

2.2.5 Crane Events Caused by Weak Program Implementation

Upon review of the 294 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 5, *Program Implementation weaknesses associated with crane issues (1968-1999)*, shows the distribution of causes for the crane issue being reported. “Not Following Procedures” was the largest category with 108. Other categories that are similar to “Not Following Procedures” would be “Ventilation” (generally failure to establish the required ventilation prior to load movements in certain areas), “Didn’t Test” (generally failure to perform crane surveillance tests prior to use) and “Load Path” (generally the failure to move loads over established safe load path areas).

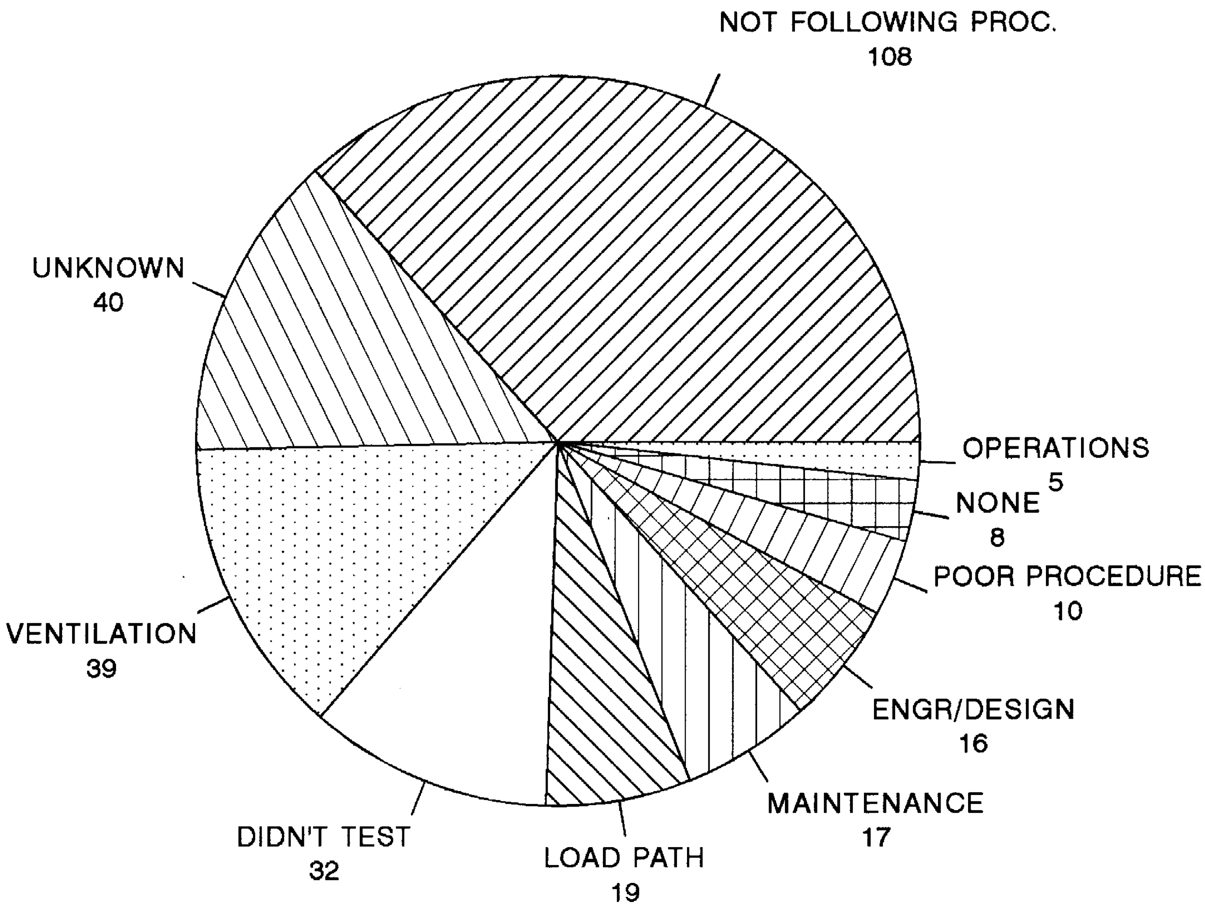

Figure 5: Program Implementation weaknesses associated with crane issues (1968-1999)

2.2.6 Safety Implication of Event

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) Loss or Partial Loss (P-Loss) of Power, (4) Load Slip, (5) Load Drop, (6) Equipment (i.e., equipment damage or failure), and (7) None (i.e., no impact on plant equipment, workers or the public). Since a load drop could also result in damage to plant equipment, more than one category could be affected. Consequently of the 294 crane issues, the total number of “outcomes” total 320. Figure 6, *Number of crane events by category (1968-1999)* indicates the number of crane issues or events for each category, and not the quantity of items affected for each event. Consequently, one crane event may result in more than one piece of equipment being damaged, or more than one death or injury.

As shown in Figure 6, over half of the crane issues resulted in no impact to plant equipment, workers or the public. In addition, most equipment damage was minor. Section 2.2.10 provides crane event details for those events which resulted in a death.

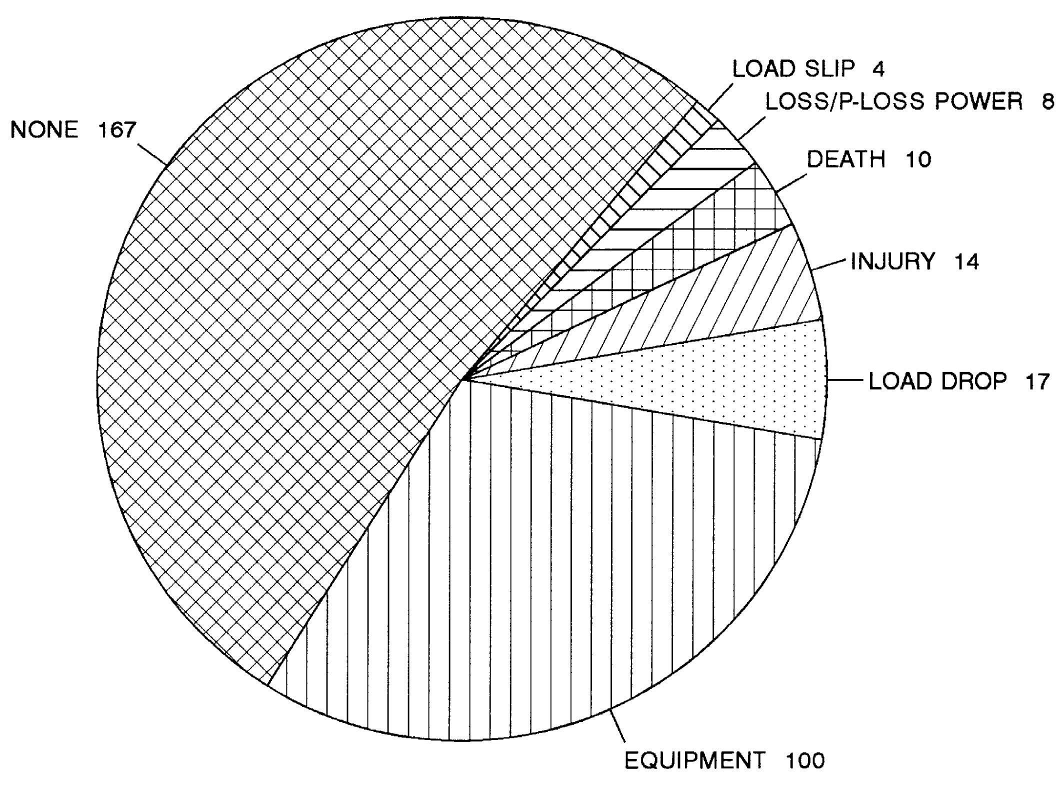

Figure 6: Number of crane events by category (1968-1999)

2.2.7 Crane Type Involved in Load Slip or Drop

During the period 1968-1999, there were 17 reported events involving a crane load drop, and 4 involving a load slip. Figure 7, *Crane type involved in load slip or drop (1968-1999)*, shows the crane type involved in load slips and load drops. In addition, Table 2, *Reported crane events involving a load drop or load slip*, provides description of each event.

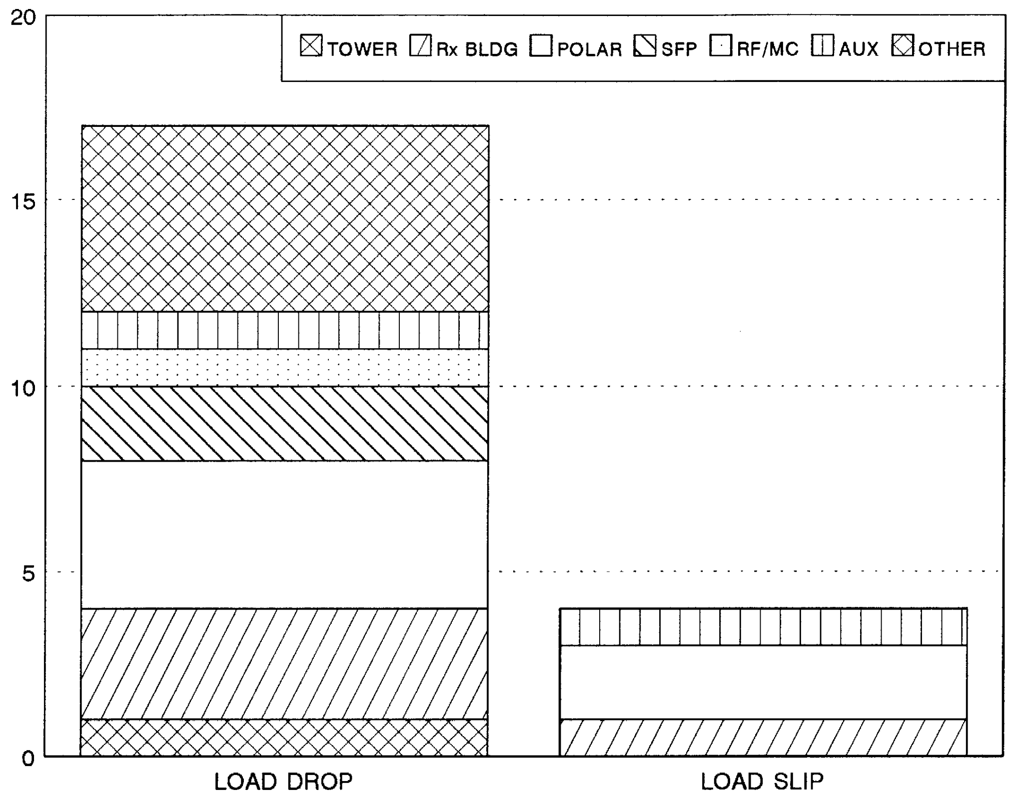

Figure 7: Crane type involved in load slip or drop (1968-1999)

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

Plant	Event Date	Load Drop/Slip	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 90 tons. The assembly was partially supported during its fall by the crane brake. The assembly tilted slightly as it fell approximately 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.
Palisades	Sept. 1970	Load Drop	A cable on a 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 approximately 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 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 1/4" bolt in the shackle holding the subassembly to fail. As a result, the subassembly fell 27 feet into the transfer tank.
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 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 as tagged out of service.
River Bend	Mar. 1983	Load Drop	A 400 ton 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 1.5 inch thick steel containment building dome had been successfully lifted and placed on the containment building by the same crane. When the form was about 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.
Brown’s Ferry 2	Mar. 1985	Load 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 25-ton capacity hook dropped through the roof of a temporary building where the maintenance workers were located.

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

Plant	Event Date	Load Drop/Slip	Event Description
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 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 1-1/2 feet onto the top of the debris bed in the reactor vessel. The dropped load weight was 220 pounds, while the crane was rated at 2000 pounds.
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 and 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.
Quad Cities 1	Dec. 1989	Load Drop	While operations were in progress to place the new Unit 2 reload fuel into the new fuel storage vault, metal shipping containers were lad out side by side on the refuel floor in a row. The containers had their lids and end caps removed. A reactor building overhead crane operator positioned the crane over the containers, and then lowered the 9 ton hoist hook to aid in aligning the hoist over the containers. The crane operator continued to lower the hoist until the hook contacted and then partially laid over on a new fuel bundle. The fuel bundle was sent to GE for examination.
North Anna 1	Feb. 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 Drop	During the replacement of the reactor vessel head, it was inadvertently lowered too far. It contacted the head alignment pins, bending the pins and causing superficial damage to the head flange. Prior to contacting the alignment pins, it was swinging back and forth approximately 6 inches. After contacting the pins, the head apparently dropped 6-12 inches and began pivoting on one alignment pin in a 6 foot arc. One pin was bent 12-14 degrees while the other was bent approximately 5 degrees.
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.
Peach Bottom 2	Sept. 1993	Load Drop	An empty irradiated component shipping liner was suspended from an auxiliary hook of the reactor building crane via an adapter about seven feet below the surface of the spent fuel pool. It dropped approximately 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.

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

Plant	Event Date	Load Drop/Slip	Event Description
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	April 1997	Load Drop	While transporting a 4000 pound toolbox using an auxiliary hoist on the reactor building crane, a nylon sling separated. One end of the box dropped approximately eight feet striking the edge of a stored Unit 2 cavity shield plug. Routine testing of slings was found to be a weakness.
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 2" 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	Load Drop	A wire support cable for the polar crane control pendant broke and caused a 100 pounds pendant with cabling of several hundred pounds to fall about 140 feet, nearly missing personnel. No cause was given.
Davis-Besse	April 1998	Load 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 200 feet into the shallow end of the refueling canal missing personnel by 3 feet.
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.
Comanche Peak 1	Oct. 1999	Load Slip	During the removal of reactor coolant pump motor 1-03, the electric hoist/chain fall failed. The 45 ton hoist was attached to the polar crane. When the hoist failed, the reactor coolant pump motor dropped approximately 15-20 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.

2.2.8 Distribution of Load Slips and Drops (1968-1999)

Figure 8, *Distribution of load slips and drops* (1968-1999), shows both the number of reported load drops and load slips, and a plot showing the number of licensed power plants for the period 1968 through 1999. As shown by the figure, there has been a slight increase in the number of both load slips and load drops over an approximate three decade period, however, this increase is substantially less than the increase in the number of licensed nuclear power plants. This trend would indicate that crane performance has greatly improved.

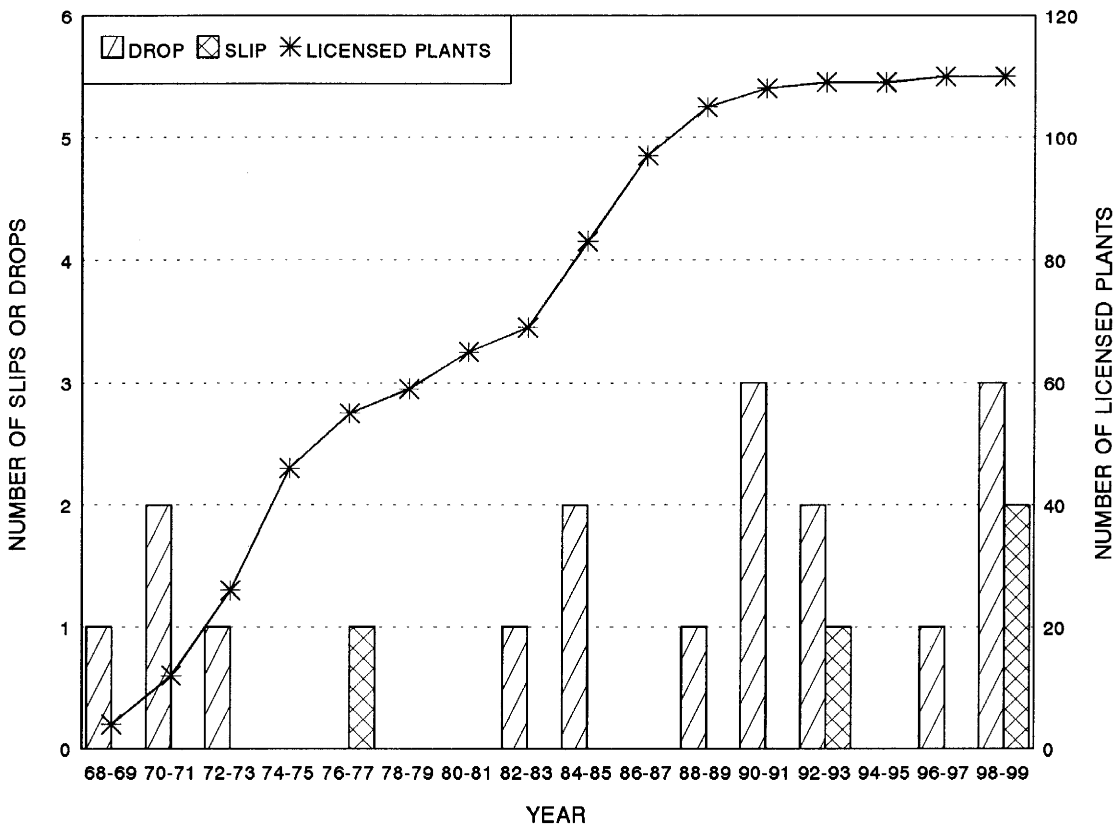

Figure 8: Distribution of load slips and drops (1968-1999)

2.2.9 Crane Events Resulting in Deaths or Injuries

Figure 9, *Crane types involved in deaths or injuries (1968-1999)*, 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.

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

- 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

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

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

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

- Auxiliary: Any of several lower capacity cranes or hoists.

As shown by Figure 9, most deaths and injuries occurred while using cranes that don't lift heavy loads near 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.

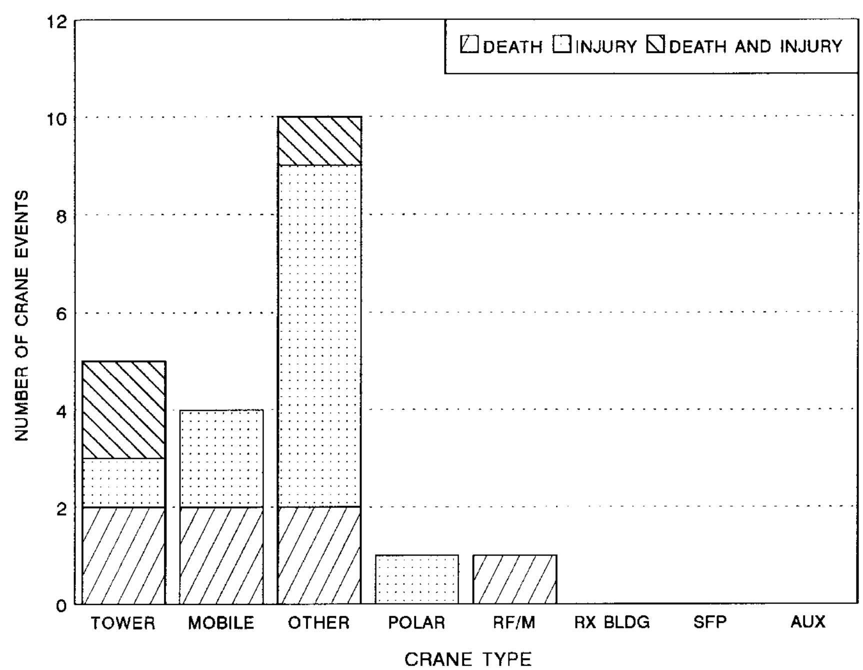

Figure 9: Crane types involved in deaths or injuries (1968-1999)

2.2.10 Description and Distribution of Crane Related Deaths and Injuries

Figure 10, *Distribution of crane related deaths and injuries (1968-1999)*, shows the number of crane related events leading to either a death or an injury. There have been 10 reported crane events that have led to deaths in the nuclear industry for the period 1968 through 1999. The highest concentration of crane related deaths and injuries at nuclear power plants occurred between 1976 and 1985. The last death in a crane related accident was 1985. Table 3, *Reported crane events resulting in deaths* provides information for each of the reported crane events that involved a death. Figure 10 also shows the cumulative number of nuclear power plants that had an operating license during the period from 1968 through 1999.

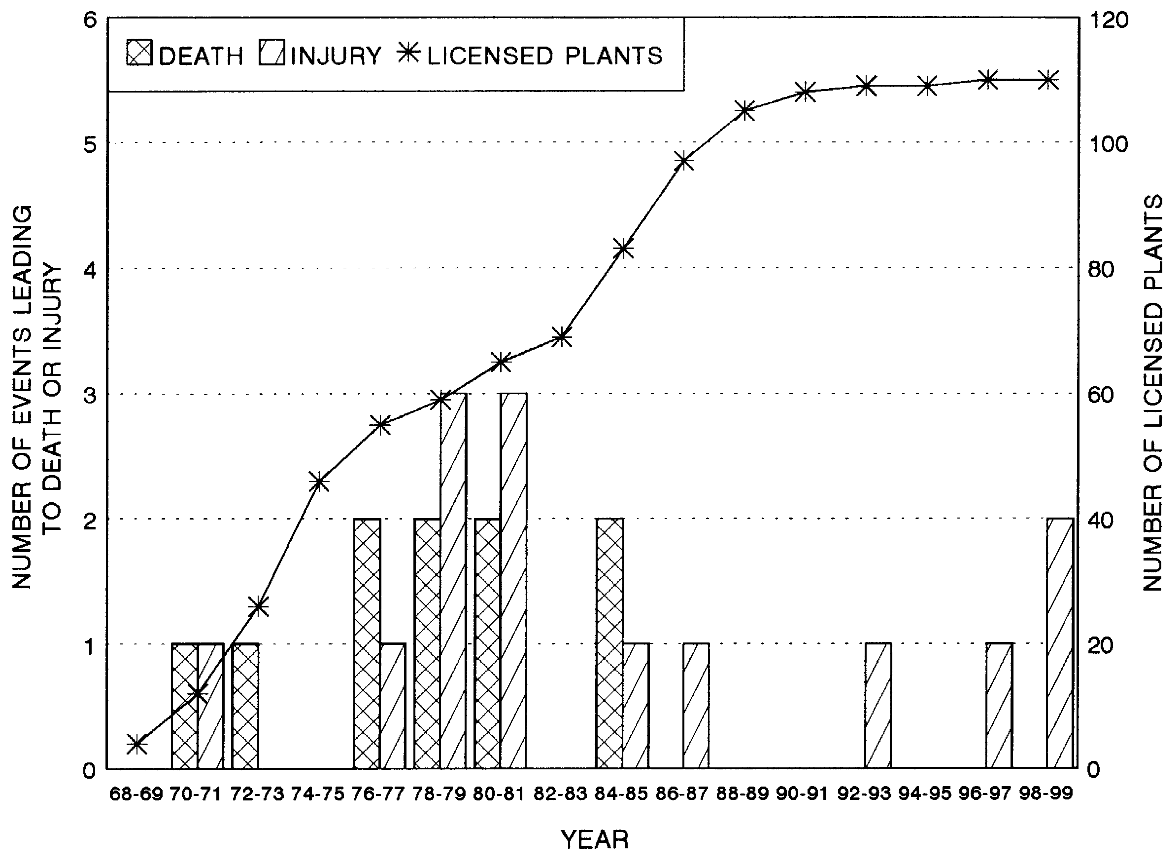

Figure 10: Distribution of crane related deaths and injuries (1968-1999)

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 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	December 1973	A worker died following a 10 feet fall from an overhead yard crane.
Peach Bottom 2,3	May 1976	A contractor employee fell 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	February 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	October 1979	A worker was killed when he touched a crane which was in contact with a high voltage overhead line.
Marble Hill 1,2	February 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	August 1980	A worker was killed when he was caught between a crane counterweight and the engine housing.
McGuire 2	February 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. The 25-ton capacity hook dropped through the roof of a temporary building where the maintenance workers were located.

2.2.11 Distribution of Crane Events by Plant

Figure 11, *Distribution of crane issues by facility, on a per unit basis (1968-1999)*, shows the number of crane issues documented against each nuclear power plant facility, divided by the number of units (i.e. units that received an operating license, or were substantially completed) at that facility. Since there are many facilities that had units canceled, judgement was used in determining how many plants were "substantially" completed, but did not receive an operating license. Four nuclear facilities reported no crane events; Hope Creek, Kewaunee, Waterford, and Watts Bar.

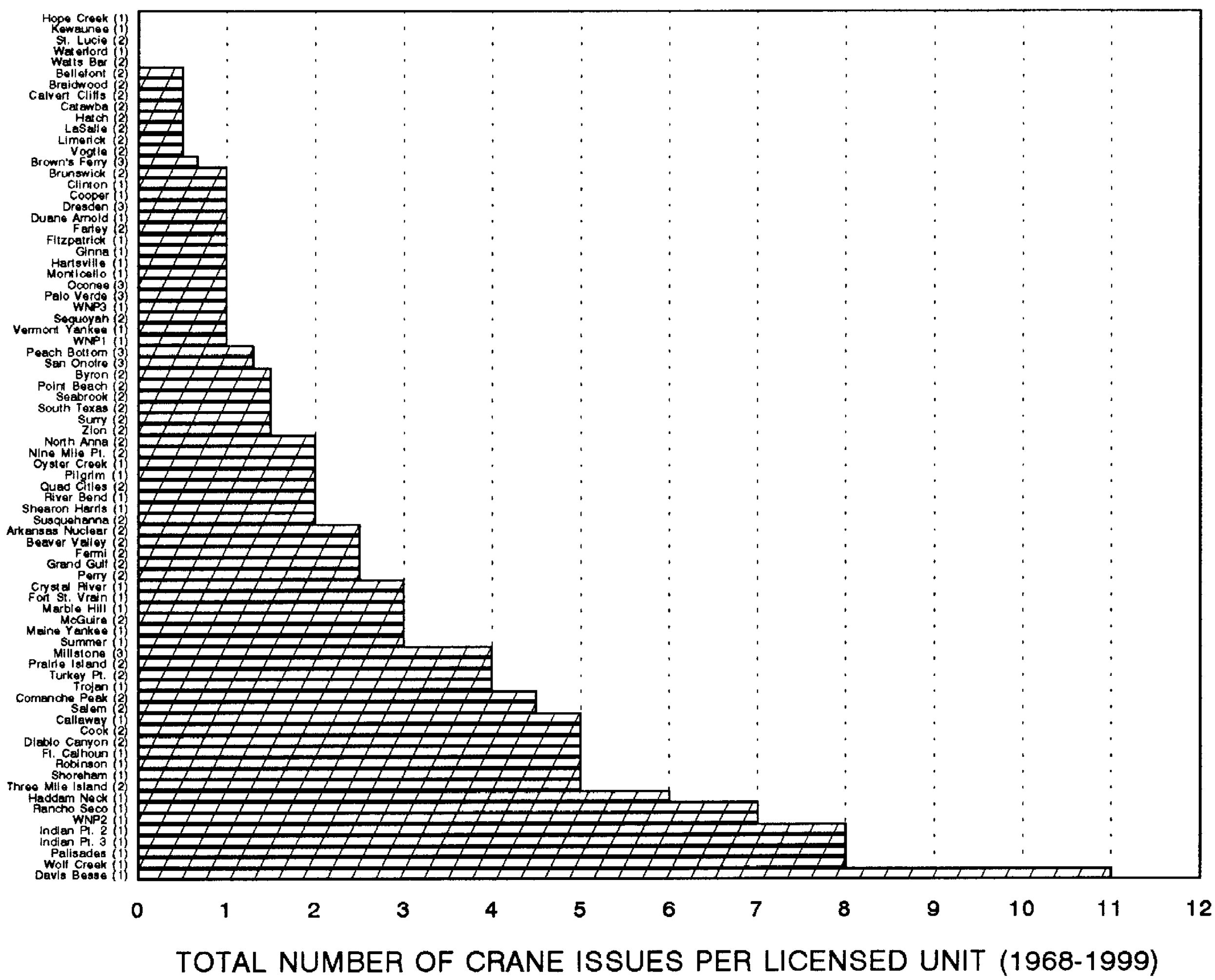

Figure 11: Distribution of crane issues by facility, on a per unit basis (1968-1999)

3 LICENSEE CRANE OPERATING EXPERIENCE SINCE COMMERCIAL OPERATION

At an initial meeting in May 2000, the Reactor Generic Issue Review Panel decided that the generic issue scope should be limited to (1) loads of approximately 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. This data was put into a database, and based on the sample, estimations of the number of very heavy loads lifts was made. Crane issues or events information was obtained from searching the NUDOCS files was used to form the basis for the industry operating experience.

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 October 1999, US nuclear power plants have operated for a combined time of approximately 1920 years. The combined operational period for the nine facilities that were visited was approximately 276 years, which is approximately a 14 percent sample. The crane operating experience sample included plants of varying designs and ages. Most were multiunit facilities, allowing more lift data to be retrieved. Table 4, *Pilot plants for crane program and operational experience reviews*, lists the facilities visited.

Table 4: 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
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 Crane Operating Experience at Pilot Plants

Table 5, *Total number of lifts with loads of approximately 30 tons or greater*, 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. 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 5: Total number of lifts with loads of approximately 30 tons or greater

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 30 tons for all US nuclear power plants, it was necessary to normalize Table 5 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. The total number of estimated very heavy load lifts for all US nuclear power plants that operated from 1980 through October 1999 was approximately 47400.

3.4 Load Slips and Drops

Of the estimated 47400 lifts, there were three "load slips" or "load drops" that involved very heavy loads. A load slip is defined as a situation where the load may descend uncontrollably, but come to a stop without impacting or damaging other equipment. A load drop is defined as a situation where the load may descend uncontrollably, but impacts other equipment and does damage. None of the three very heavy load events resulted in radiation releases, risks to licensee personnel, or the public, as shown in Table 6, *Load slips and load drops occurring at operating nuclear facilities (1980-1999)*.

Table 6: Load slips and load drops occurring at operating nuclear facilities (1980-1999)

Plant	Event Date	Load Slip	Load Drop
Fort Calhoun	May 1990		While lowering the reactor head, it cocked slightly, catching on alignment pins, bending two.
Arkansas Nuclear One-1	November 1993	When removing the reactor head, the head was trolleyed horizontally. When a vertical lift was attempted, the head instead lowered.	
Comanche Peak 1	October 1999	A gearbox in an auxiliary hoist (attached to the polar crane) failed, lowering the reactor coolant pump motor about 15-20 feet. It came to rest before impacting any equipment.	

3.5 No Accident Sequence Precursor Events Involving Cranes

A review of Accident Sequence Precursor (ASP) data for the period 1985 through 1999 was performed to determine if any crane related event met the thresholds for risk significance. The ASP program identifies and categorizes precursors to potential sever core damage accident sequences. Accident sequences are those that, if additional failures occurred, could have resulted in inadequate core cooling, causing sever 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 1.0×10^{-6}.

The most potentially risk-significant crane events involved loss or partial loss of offsite power. For the period 1985 through 1999, there were six such mobile crane events. These are summarized in Table 7.

Table 7: Crane events resulting in a loss or partial loss of offsite power

Plant	Event Date	Description
Peach Bottom 2	August 1987	While Unit 2 had been shutdown for five months, an 80 ton 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)
Fermi 2	December 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. (No LER was written)
Palo Verde 3	November 1991	While Unit 3 was in hot standby, a 35 ton 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)
Diablo Canyon 1	March 1991	While Unit 1 was in a refueling outage, loss of offsite power caused by mobile crane when it got too close to a 500 KV electrical line. The 230 KV startup power system had been cleared for maintenance and was not available. RHR capability was lost for less than one minute, and the spent fuel pool pumps were inoperable for approximately 23 minutes. An Unusual Event was declared. (LER 275-91-004-01, also an AIDED inspection was performed)
Nine Mile Point 2	September 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. Emergency power was provided by two EDGs. (LER 286-95-004)

Of the six crane events described in Table 7, two licensees had Augmented Inspection Team (AIT) inspections (Palo Verde and Diablo Canyon). However, none of the six mobile crane events met the minimum risk threshold requirements to be classified as an ASP event.

3.6 Crane Event Tree and Potential Consequences

BWR plants are more risk significant, given the occurrence of some heavy load drops than PWRs because of the location of the spent fuel pool on the upper floor of the reactor building, and the heavy loads that the refueling floor would experience. 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. Should a load drop occur while the load is being lowered down the equipment hatch to ground level from the refueling floor (approximately 100 feet), the torus could be punctured. Accident mitigation could be compromised given a punctured torus (emergency core cooling system pump failure) or during suppression pool cooling. A heavy load drop that would penetrate the refueling floor could also disable an isolation condenser (installed at some BWRs) which would also compromise the plant's capability to cope with decay heat removal following a station blackout. Other scenarios exist where individual trains of safety-related systems could be disabled, but not to the point where system redundancy or diversity would be eliminated.

Because of the vast differences between reactor safety system layout even within the same design type (i.e., BWR vs. PWR, or NSSS vendor) more exact consequence analysis of very heavy load drops at different locations within a nuclear plant is not practical. 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 9. Of the 74 facilities listed on Table 9, only 8 licensee responses to Bulletin 96-02 indicated that a consequence analysis had been done at their facility for heavy load drops.

When taken together, the overall probability of a very heavy load drop with significant consequences is very low. A generic event tree was developed (see Figure 12, *Load drop event tree*) assuming that the load drop was the initiating event. Probabilities for each branch were conservatively estimated using information gathered from the Pilot Plant licensees, and NUREG-0612.

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

The number of very heavy load lifts per reactor year (25) was calculated by taking the total number of very heavy load lifts (47400 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 (1920 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, there were <u>no load drops</u> of any consequence. To be conservative, one load drop was assumed to occur during the period of interest. Assuming that the number of very heavy load lifts was approximately 47400, the load drop frequency (drops/number of lifts) was calculated to be approximately $2x10^{-5}$ (1/47400 lifts).

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

The probability of a drop over an Safe Shutdown Equipment (SSE) would be related to the probability of the failure to follow procedures. As shown in Figure 5, a large percentage of crane issues are either related to not following procedures, or not properly implementing procedures. For the purposes of this assessment, it was conservatively assumed that all failures to follow procedures (108) event could have caused a drop over an SSE. This would result in a probability of 108/47400 or approximately 2E-03 failures per lift. A much more conservative probability value was given in NUREG-0612 which was referenced as coming from WASH-1400, Reactor Safety Study, of 1E-02. This study assumes a value 10 times the WASH-1400 value, i.e., 1E-01.

Safe Shutdown Equipment Needed (On Level)

NUREG-0612 conservatively estimates other event probabilities relating to failure to maintain systems per design ranging between $1x10^{-1}$ and $5x10^{-1}$ per event. This value would be a very conservative estimation of an SSE being needed to mitigate an accident because of separation of trains and redundancy. Similar very conservative probabilities are also used for floor breach because of the potential common mode or common cause influences.

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 12 that would involve a floor breach are different. A factor of 10 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. 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 100 from the “best case” to the “worst case.”

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 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 an SSE exists below level was conservatively assumed to range between 1E-01 and 8E-01. The higher probability value (8E-01) 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 Needed (Below Level)

Transporting very heavy loads over equipment that would be necessary for plant accident mitigation would not be a conservative practice, 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 1E-02 and 5E-02. For the purposes of this study, the probability is assumed to be increased by a factor of 10, or 1E-01 and 5E-01, to compensate for potential common cause failures due to other preceding failures in the same pathway.

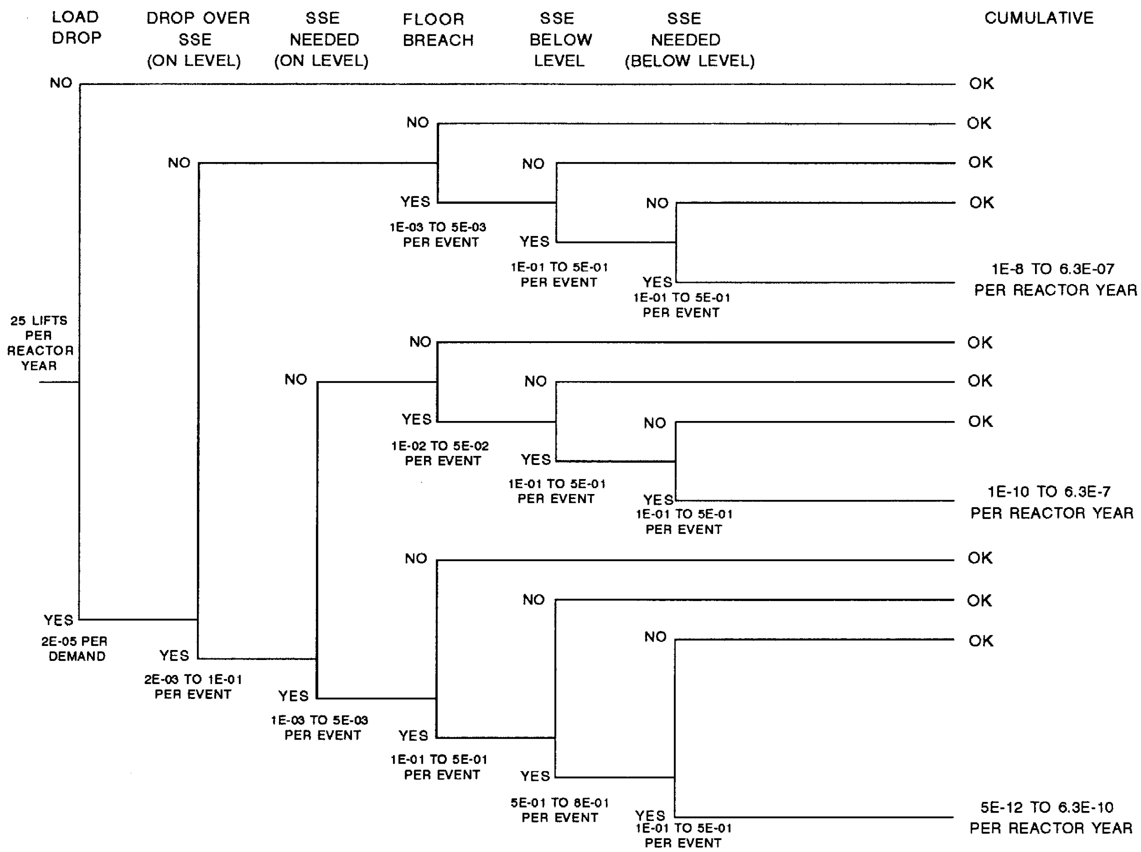

Figure 12: Load drop event tree

4 LICENSEE LOAD DROP CALCULATIONS VARIED GREATLY

A sampling of load drop calculations obtained from each facility that was visited indicated that calculational methodologies and assumptions varied greatly from licensee to licensee, producing radically different end results. Heights of load drops, plant locations for postulated load drops, contact area at impact, materials property values, and weights of loads varied greatly. The Oyster Creek calculation for a drop of a 45 ton fuel cask over a reinforced concrete 16 inch thick slab was the most restrictive, with an allowable drop height of 2.77 inches. Some facilities performed load drop calculations using equations that were intended for ballistic type situations meant for high velocity and low mass. 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 8, *Heavy load drop calculations*, provides a sampling of load drop calculations from the facilities that were visited.

Table 8: Heavy load drop calculations

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Grand Gulf	8/15/78 Bechtel	Drywell head	61.5	5ft (air)	- Refueling floor; 9" RC slab on 3" decking (non-composite), slab supported on W36x300 beams @ 6'4" spacing	17.9 ft/sec	Used an equation for penetration of 12" diameter missiles. 100% of flange will contact the floor.	- Depth of penetration 2.8" - 9" RC slab μ = 6.9 - W36x300 μ = 5.9
Grand Gulf	8/17/78 Bechtel	Drywell head	61.5	30 ft (air)	- Reactor well; 1.5" wide sleeve, radius of 16'-3/4"	43.9 ft/sec	Drywell head hits the sleeve	Drywell head crushes the sleeve, and continues downward, but doesn't compromise the integrity of the RPV
Grand Gulf	8/16/78 Bechtel	RPV head	117	5 ft (air)	- Refueling floor, 4-'0" thick RC	17.9 ft/sec	100% of flange will contact the floor.	-Depth of penetration 4.4" -For simple support, μ=9; for fixed support, $\mu<1$
Grand Gulf	4/4/78 Bechtel	Steam separator	68	17 ft (water)	- Spent fuel pool; Steam separator area, 52" thick slab with 1/4" liner plate	21.5 ft/sec	Steam separator falls in water	-Assuming a 1/4" plate, the depth of penetration = .7" (unsatisfactory) -Assuming a 52" concrete slab, depth of penetration = 6.2" -Assuming an interface forcing function, depth of penetration = 2.6" -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	40	23 ft (air)	- Dryer storage area, 52" thick slab with 1/4" liner plate	38.5 ft/sec	- For the 1/4" liner plate, the equation appears to spread out the load over an entire cylinder with a diameter of 238"(same for the slab) as opposed to an annulus.	- Assuming a 1/4' plate, the depth of penetration = .09" - Assuming a 52" thick concrete slab, depth of penetration = 5.4" - $\mu<5.3$
Oyster Creek	10/29/99 EQE	Fuel cask	45	6" (air)	- Refueling floor; At the center of beam 5B27; slab thickness 16"; beam width 36," beam depth 30"; various rebar 8-15, #8	5.7 ft/sec	- ACI 349-97	- Allowable drop height = 7.01"

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oyster Creek	10/26/99 EQE	Fuel cask	45	(3.85") (air)	- Refueling floor; center drop on slab 5S10; slab span N/S 23'-3" x E/W 20'-9"; slab thickness 16"; rebar #6 @7" and 18" centers, and #8@6, 8, &9" centers;	(4.55 ft/sec)	- ACI 349-97	- Allowable drop height = 3.85"
Oyster Creek	10/26/99 EQE	Fuel cask	45	(2.77") (air)	- Refueling floor; Drop on slab 5S10 adjacent to beam 5B27; slab span N/S 23'-3" x E/W 20'-9"; slab thickness 16"; rebar #6 @7" and 18" centers, and #8@6, 8, &9" centers	(3.86 ft/sec)	- ACI 349-97	- Allowable drop height = 2.77"
Oyster Creek	10/26/99 EQE	Fuel cask	45	(11.58") (air)	- Refueling floor; Drop on slab 5S14 adjacent to beam 5B39; similar to slab 5S10 but slab thickness = 26";	(7.88 ft/sec)	- ACI 349-97	- Allowable drop height = 11.58"
Oyster Creek	10/26/99 EQE	Fuel cask	45	6" (air)	- Refueling floor; Drop on east wall of spent fuel pool; the wall is 6' thick and extends from the 119' level to the 72' level;	5.7 ft/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; target mass 10000 lb/ftsec^2	- Available strain energy calculated at ~254 kipft, kinetic energy from drop ~31 kipft - If kinetic energy of drop is set equal to the strain energy, the allowable drop height would be 49.6" - If load is dropped directly on C6, the allowable drop height would be 49"
Palo Verde	6/4/80 Bechtel	Fuel cask	125	12' (air)	- Drop from level 124.5' to the decontamination pit (~12');	27.8 ft/sec	- Assumes that the cask hits the floor exactly flat; Ductility ratio of 30 acceptable	- Thickness required to preclude spalling 71.56"; slab defection .063"; ductility ratio calculated to be 22.84

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Palo Verde	6/4/80 Bechtel	Fuel cask	125	30' (air)	- Drop from the top of the spent fuel pool to the bottom of the cask loading pit; target slab is 7'-9" thick	44.96 ft/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	126	1' + rotation strike on wall	- Drop from top of spent fuel pool to the decontamination pit and then deflects to the east wall of the pit	(Striking velocity on the wall = 16.133 ft/sec)	- Ductility ratio of 30 acceptable	- Calculated ductility ratio 47.09, 20 (average of beam, 10, and slab 30) - For this situation, an energy absorbing pad was required
Brown's Ferry	1/14/72 TVA	Fuel cask	100	3'	- Drop on hypothetical 18" RC slab	13.9 ft/sec	- NAVDOCKS (p51) - Cask lands flat on 16 fins, evenly distributed (4.124 ft^{2})	- Depth of penetration = .0892 ft
Brown's Ferry	1/17/72 TVA	Fuel cask	100	3'	- Drop on hypothetical 18" RC slab	13.9 ft/sec	- Compares energy absorbed to the energy the system can ultimately absorb	- energy to be absorbed = 7.2 E6, in-# energy the system can ultimately absorb = 9.35 E6 in-#
Brown's Ferry	1/18/72 TVA	Fuel cask	100	3'	- Drop on 18" thick slab near supports	13.9 ft/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, it will not go through the slab
Brown's Ferry	1/27/72 TVA	Fuel cask	100	6"	- Drop on 36" slab	5.675 ft/sec	- Uses a modified Petry formula for penetration	- Penetration calculated to be .015 ft
Limerick	4/30/84 Bechtel (1)	Drywell head	104	3'	- Tilted drop on refueling floor, RC 24" thick, #9@8" centers (T&B);	13.9 ft/sec	- Capacity of slab based on yield-line theory, simple span, elasto-plastic design - Doesn't appear to account for kinetic energy absorption over a small area - Tilted drop case, strikes over 40 degrees of circumference - Interface force = 6.35 E6 # (average=2.1 E6 #)	- Punching shear capacity appears to be high (240 psi) - Calculated punching shear appears to be low (117psi) - Compressive strength of concrete appears to be high - E for concrete appears to be high - μ = .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)

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/24/84 Bechtel (2)	Drywell head	104	3'	- Flat drop on refueling floor	13.9 ft/sec	- Drywell head lands completely flat on the refueling floor - Interface force = 7.1 E6 #	- 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 8.72 (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	92	3'	- Flat drop on refueling floor	13.9 ft/sec	- RPV head lands completely flat on the refueling floor - Interface force = 1.23 E7 #	- 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	92	3'	- Tilted drop on refueling floor	13.9 ft/sec	- RPV head lands tilted - Interface force = 7.16 E6 # (average=2.39 E6 #)	- 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	92	2'	- Tilted drop on refueling floor	11.38 ft/sec	- RPV head lands tilted - Interface force = 6.55 E6 # (average=2.18 E6 #)	- μ ~ 20, (over W24, Zone C) - Drop height was changed from 3' to 2' to get a lower μ
Limerick	4/26/84 Bechtel (6)	Shield Plugs	12	3'	- Flat drop on refueling floor	13.9 ft/sec	- Flat drop calculated for over W36, 24" thick concrete, and W24 - Interface force = 3.37 E7 #	- Flat drop force for the 12 ton plugs was calculated to be greater than the tilted drop of the drywell head at 104 tons - μ = 1.5, allowable 10, (over concrete, Zones A&B) - $\mu \ll$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	59	3'	- Flat drop on refueling floor	13.9 ft/sec	- Flat drop - Contact area = 75 ft^2 - Interface force = 5.3 E7 #	- μ = 2, allowable 10, (over concrete, Zones A&B) - μ = 1.08, allowable 8.72 (over W36, zones A&B) - $\mu <$2.5, allowable 10, (over concrete, Zone C) - μ = 1.53, allowable 10, (over W24, zone C)

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/26/84 Bechtel (8)	Stoplog	59	3'	- Tilted drop (45 degrees) on refueling floor	13.9 ft/sec	- Tilted drop - Contact area = 2.5 ft^2 - Interface force = 1.78 E6 # (average=5.94 E5 #)	- Per an unreferenced equation, spalling of concrete will occur at a drop height of approximately 4" - μ = .6, allowable 10, (over concrete, Zones A&B) - Punching shear capacity appears to be high (240 psi from p. 12 of calc) - Calculated punching shear appears to be low (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	59	1'-9"		10.6 ft/sec	- Tilted drop - Contact area = 2.12 ft - Interface force = 1.14 E6 # (3.814 E5 #)	- $\mu \sim 20$, (over concrete with embedded beams)
Limerick	4/26/84 Bechtel (10)	Stoplog	38	2'	- Tilted drop (45 degrees) on refueling floor	11.35 ft/sec	- Tilted drop - Contact area = 1.3 ft^2 - Interface force = 9.41 E5 # (average=3.134 E5 #)	- Per an unreferenced equation, spalling of concrete will occur at a drop height of approximately 7" - $\mu < 1.0$, allowable 10, (over concrete, Zones A&B) - Punching shear capacity appears to be high (240 psi from p. 12 of calc) - Calculated punching shear appears to be low (101 psi) - μ = 1.2, allowable 8.72, (over W36, zones A&B) - $\mu \ll 1.0$, (over concrete, Zone C) - $\mu \sim 12$, allowable 12, (over W24, zone C)
Limerick	4/26/84 Bechtel (11)	Stoplog	38	2'	- Flat drop on refueling floor	11.35 ft/sec	- Flat drop - Contact area = 135 ft^2 - Interface force = 8.68 E6 #	- 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 16.1" - μ = 1.5, allowable 10, (over concrete, Zone C) - $\mu < 12$, allowable 12, (over W24, zone C)
Limerick	4/26/84 Bechtel (12)	Stoplog	38	1.5' (air) 22.5' (water)	- Flat drop back into its slot	30.9 ft/sec	- Flat drop - assume 50% contact (831.25 in^2) - Interface force 2.73 E7 # (average=9.1 E6 #)	- Penetration based on impact duration = 1.4" - Penetration based on missiles hitting soils = .68"

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/26/84 Bechtel (13)	Stoplog	38	1.83' (air) 37.75' (water)	- Flat drop into the Fuel Pool	34.5 ft/sec	- Flat drop - Assumes 50% contact (831.25 in^2) - Interface force 3.04 E7 # (average=1.01 E7 #)	- Penetration based on impact duration = 1.7" - Penetration based on missiles hitting soils = .85"
Limerick	4/23/84 Bechtel (14)	Steam dryer assembly	45	6'	- Flat drop on refueling floor	19.7 ft/sec	- Flat drop - Total contact area = 3000 in^2 - Contact area for slab of interest = 1140 in^2	- 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	45	6'	- Tilted drop (17.46 degrees) on refueling floor	19.7 ft/sec	- Tilted drop - Contact area = 4.06 ft^2 - Interface force = 4.07 E6 # (average=1.36 E6 #)	- $\mu < 1$, 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	45	5'	- Tilted drop (14.5 degrees) on refueling floor	17.94 ft/sec	- Tilted drop - Contact area = 4.18 ft^2 - Interface force = 3.83 E6 # (average=1.28 E6 #)	- 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	45	3'	- Tilted drop (8.62 degrees) on refueling floor	13.9 ft/sec	- Tilted drop - Contact area = 7.29 ft^2 - Interface force = 5.17 E6 # (average=1.72 E6 #)	- μ = 12, allowable 12, (over W24, zone C)
Limerick	4/28/84 Bechtel (18)	Steam separator assembly	81.5	5'	- Flat drop on refueling floor	17.9 ft/sec	- Flat drop - Contact area = 5.61 ft^2 - Interface force = 5.12 E6 #	- 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	81.5	5'	- Tilted drop (14.5 degrees) on refueling floor	17.9 ft/sec	- Tilted drop - Contact area = 4.97 ft^2 - Interface force = 4.55 E6 # (average=1.52 E6 #)	- $\mu < 1$, 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	81.5	2.5'	- Tilted drop (7.2 degrees) on refueling floor	12.7 ft/sec	- Tilted drop - Contact area = 5.57 ft^2 - Interface force = 3.595 E6 # (average=1.2 E6 #)	- μ = 12, allowable 20 (two beams, over W24, zone C)

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Limerick	4/28/84 Bechtel (21)	Steam separator assembly	81.5	7'	- Flat drop on refueling floor	21.2 ft/sec	- Flat drop - Contact area on slab of interest = 7.92 ft^2 weight on slab of interest = 31 tons - Interface force = 8.56 E6 #	- μ = 3.5, allowable 10, (over concrete, Zones A&B) - μ = 2.8, allowable 10 (over W36, zones A&B)
Limerick	4/28/84 Bechtel (22)	Steam separator assembly	81.5	7'	- Tilted drop (20.5 degrees) on refueling floor	21.2 ft/sec	- Tilted drop - Contact area = 4.88 ft^2 - Interface force = 5.27 E6 # (average=1.76 E6 #)	- μ = .7, allowable 10 (over W36, zone D)
Limerick	5/8/84 Bechtel (23)	Shield plugs	85	3'	- Flat drop on refueling floor	13.9 ft/sec	- Flat drop - Total contact area = 418.5 ft^2 - Contact area on slab of interest = 181.9 ft^2 - Interface force = 1.29 E8 #	- $\mu < 10$, allowable 10, (over concrete, Zones A&B)
Limerick	5/8/84 Bechtel (24)	Shield plugs	85	3'	- Tilted drop (5.3 degrees) on refueling floor	13.9 ft/sec	- Tilted drop - Total contact area = 2.65 ft^2 - Interface force = 1.88 E6 # (average=6.27 E5 #)	- $\mu < 1$, 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	85	2'	- Tilted drop (3.5 degrees) on refueling floor	11.3 ft/sec	- Tilted drop - Total contact area = 2.67 ft^2 - Interface force = 1..54 E6 # (average=5.1 E5 #)	- μ = 12, allowable 12 (two beams, over W24, zone C)
Limerick	6/17/96 S&L	Shield plug	85		- Tilted blunt drop on drywell head		- Slightly tilted drop - Drywell head materia l thickness at impact is 1.5" 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 % of load hits the drywell head - S&L doesn't provide an analysis for a sharp (small area) impact - Area of impact = 754 in^2	- Once the effective load of the plug is reduced from 170 k# to 79 k#, the strain energy to be absorbed by the drywell head was calculated to be 3402 in-kips - Increased the capability of the head toy a DIF of 1.2, the materials can take 4774 in-kips which is about 40% higher than that caused by the plug drop - The deflection at maximum strain energy would be approximately 8", whereas at the calculate strain energy, the drywell head will deflect approximately 5.8"

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Comanche Peak	12/8/88 SWEC (4)	Reactor Coolant Pump Assembly	27.6		- 20 " thick RC - Slabs S-4 to S-8		- (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" = 7 ft. Diameter	- Maximum drop height = 5" (Scabbing) - The contact areas was changed in calculation listed as 4-1 below
Comanche Peak	12/8/88 SWEC (4-1)	Reactor Coolant Pump Assembly	27.6		- 20 " thick RC - Slabs S-4 to S-8		- (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" = 2 ft. Diameter	- Maximum drop height = 1' - 2"
Comanche Peak	12/8/88 SWEC (4-2)	Reactor Coolant Pump Assembly	27.6		- 26 " thick RC - Slabs S-1, 2, 3, and 9		- (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" = 2 ft. Diameter	- Maximum drop height = 3' - 3""
Comanche Peak	12/8/88 SWEC (5-1)	Reactor Coolant Pump Stator	23.8		- 20 " thick RC - Slabs S-4 to S-8		- (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" = 6 ft. Diameter	- Maximum drop height = 5"
Comanche Peak	12/8/88 SWEC (5-2)	Reactor Coolant Pump Stator	23.8		- 26 " thick RC - Slabs S1, 2, 3, and 9		- (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" = 6 ft. Diameter	- Maximum drop height = 1' - 3"
Comanche Peak	12/8/88 SWEC (6-1)	Reactor Coolant Pump Motor Assembly (Rotor & Stator)	42.4		- 20 " thick RC - Slabs S-4 to S-8		- (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" = 2 ft. Diameter	- Maximum drop height = 9"

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Comanche Peak	12/8/88 SWEC (6-2)	Reactor Coolant Pump Motor Assembly (Rotor & Stator)	42.4		- 26 " thick RC - Slabs S1, 2, 3, and 9 - Slab S-10		- (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" = 2 ft. Diameter	- Maximum drop height = 2' - 1"
Comanche Peak	12/8/88 SWEC (7/7A)	Reactor Coolant Pump Motor Assembly (Rotor & Stator)	27.6		- 54 " thick RC - Slab S-10		- (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" = 2 ft. Diameter	- Maximum drop height = 48' - 9" (Scabbing) - Maximum drop height = 24' - 4" (to reach strain energy max)
Comanche Peak	12/8/88 SWEC (8/8A)	Reactor Coolant Pump Rotor)	3.3		- 54 " thick RC - Slab S-10		- (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" = 2 ft. Diameter	- Maximum drop height = 353' - 11" (Scabbing) - Maximum drop height = 176' - 8" (to reach strain energy max)
Oconee	6/1/82 (1)	- Low pressure turbine rotor	138	30 feet above turbine deck	- Turbine deck floor 11.5" thick RC - Second floor 8" thick RC - Base floor 48" RC	- 43.95 ft/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 10.31," 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 7" into the basement floor - Will not damage any piping greater than 14.12"
Oconee	6/1/82 (2)	- Low pressure turbine rotor	138	77 feet	- 60" thick RC basement floor	70.4 ft/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 = 21.12" - Some spalling may occur - With not prevent vital embedded systems from performing their safety related functions

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oconee	10/16/75 (3)	- Spent fuel cask	24	46.5 feet (40 feet throug h water)	- Floor of spent fuel pool	54.72 ft/sec	- Allow one trunnion or side of yoke to fail, load stabilizes, and then falls to the SFP floor - Cask hits at approx. 11 degrees - Uses modified Petry formula	- Penetration in steel floor plate 1.91 inches. Actual thickness of plate on the floor is 2.25"
Oconee	5/19/89 (4)	-Spent fuel cask	100		- Floor of spent fuel pool	55 ft/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 = 11.4 “
Oconee	5/19/89 (5)	-Spent fuel cask	100	46.5 ft	- Floor of spent fuel pool	46 ft/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 = 6.8 “
Oconee	5/26/89 (6)	- Spent fuel cask			- Floor of spent fuel pool		- Assumes that the largest crack possible would be 1/64" wide and could include the largest plate in the spent fuel pool (568" in perimeter) - Assumes that 40' of water is in the pool	- The leakage rate was calculated to be 21.3 gallons per day
Oconee	11/21/80 (7)	- Spent fuel cask	29.1	27' - 9"	- Fuel rack	42.27 ft/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)	43.4	27' - 9"	- Fuel rack	42.27 ft/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

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Oconee	2/26/88 (9)	- Radiological consequences of spent fuel cask drop in pool	NA	NA	- Fuel rack	NA	- 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 does from plume is insignificant	- Total body dose (Rem) for units 1 and 2 = .15 - Total body dose (Rem) for unit 3 = .13 - Thyroid dose (Rem) for units 1 and 2 = 72 - Thyroid dose (Rem) for unit 3 = 72
Diablo Canyon	9/16/86 Bechtel (1)	RCP motor stator	10	12"	- RC slab, infinite thickness	8.02 ft/sec	- Assumes infinite slab thickness - Assumes missile impact	- Depth of penetration = 0.038"
Diablo Canyon	9/16/86 Bechtel (1)	RCP motor stator	10	12"	- RC slab, 24" thick	8.02 ft/sec	- Assumes slab thickness of 24" - Assumes missile impact	- Depth of penetration = 0.038"
Dresden 1	9/28/93 Bechtel (1)	TN-RAM cask	38.5	2' above pool water, 41' of water	- Bottom of spent fuel pool - RC2-3 ft. thick with rock base	- 44.9 ft/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	38.5	2' above pool water, 41' of water	- Bottom of spent fuel pool - RC2-3 ft. thick with rock base	- 38.4 ft/sec	- Bechtel Design Guide C-2.45, “Design of Structures for Tornado Missile Impact, Rev 0" - Drops horizontally, contact area is calculated assuming a .76 inch penetration (1631 square inches)	- The concrete base will fail in shear
Dresden 1	9/28/93 Bechtel (3)	Tn-9.1 Cask	41.5	2' above pool water, 41' of water	- Bottom of spent fuel pool - RC2-3 ft. thick with rock base	- 38.4 ft/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

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Dresden 1	9/28/93 Bechtel (4)	Tn-9.1 Cask	41.5	2' above pool water, 41' of water	- Bottom of spent fuel pool - RC2-3 ft. thick with rock base	- 29.0 ft/sec	- Bechtel Design Guide C-2.45, "Design of Structures for Tornado Missile Impact, Rev 0" - Drops horizontally, contact area is calculated assuming a .406 inch penetration (2044 square inches)	- The concrete base will fail in shear
Dresden 1	10/6/93 Bechtel (5)	Tn-9.1 Cask	41.5	6" (this dimension should have been at least 10 inches	- 8" thick RC wall	- 5.67 ft/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 10 inches, not 6 inches as assumed in the calculation	- spalling will not occur since wall is >>31.9 inches - Speculation is made for drops on the walkway next to the transfer pool
Dresden 1	9/28/93 Bechtel (6)	TN-RAM cask	38.5	12"	- Washdown area floor - 9" thick RC slab	- 8.02 ft/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	38.5	- 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% of allowable)
Dresden 1	10/5/93 Bechtel (8)	Tn-9.1 Cask	41.5	- 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% of allowable)
Dresden 1	10/6/94 Vectra (9)	Spent fuel casks	75-110	- 3.75 ft. air, 39.25 water	- Fuel transfer slab, RC 3ft. thick	- Variable from approximately 38 to almost 47 ft/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% contact) for all equations - Punching shear is the controlling failure mode	- Acceptable for penetration, perforation and spalling (however, impact area of 100% 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 17 ft. x 10 ft. (Even assuming a flat cask impact area)

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Dresden 1	10/6/94 Vectra (10)	Spent fuel cask	110	NA	- Fuel transfer slab, RC 3ft. thick	- approximately 40-42 ft/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 154 square ft - A coefficient of permeability (.0137 ft/day) for a sandy clay soil will be assumed	- Maximum leakage calculated to be approximately 2.7 gal/minute which should be easily made up by available water sources
Dresden 2,3	5/21/73 S&L (11)	IF-300 GE cask	70	1.88 ft in air, 37.75 ft in water	- Spent fuel pool floor - 6'-3" RC slab	- 44.1 ft/sec	- Vertical drop - Modified Petry formula - ACI 318-71 - Assumes a flat cask impact (100% impact area of the fins, 445.5 square inches)	- Penetration in slab = 10.03" - Load factor against punching shear = 2 - Lad factor against cracking = 1.44 -
Dresden 2,3	5/21/73 S&L (12)	IF-300 GE cask	70	1.88 ft in air, 37.75 ft in water	- Spent fuel pool floor - 6'-3" RC slab	- 43.9 ft/sec	- Horizontal drop - Modified Petry formula - ACI 318-71 - Assumes a reduce contact area of 1008 square inches	- Penetration in slab = 4.5" - Load factor = 1.5 - Load factor against punching shear = 2
Dresden 2,3	5/21/73 S&L (13)	IF-300 GE cask	70	NA	- Decontamination pit	NA	- Vertical drop - ACI 318-71 - Due to the complex shape, the slab was transformed into an equivalent fixed ended beam of 9.5' width	- The maximum drop height was calculated to be 11.15 inches - It was recommended that the cask be raised a maximum of 9" for safe cleaning operation, and 6" while traveling to and from the decontamination pit
Dresden 2,3	5/21/73 S&L (14)	IF-300 GE cask	70	NA	- Travel path between the decontamination pit and the spent fuel pool over the torus	NA	- An extension form (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 22." - A conservative lift height of 6" was made
Dresden 2,3	7/2/81 S&L (15)	NA	NA	95.5 ft	- Drop down the reactor building equipment hatch to the main floor over the torus	- 78.4 ft/sec	- Assume the dropped load has a diameter of 18" - RC slab, 24 inches thick - Assumes concrete will fail at approximately 1300 Kips, then calculates the penetration depth into the concrete from an initial height of 95.5 ft	- To prevent scabbing of a 24" thick floor, the missile penetration depth cannot be > 3.27" - Maximum load drop with no scabbing of a 24" thick slab = 1 ton

PLANT	CALC DATE	LOAD	WT (tons)	HT	TARGET COMPOSITION	STRIKING VELOCITY	ASSUMPTIONS	RESULTS
Dresden 2,3	7/2/81 S&L (16)	NA	NA	95.5 ft	- Drop down the reactor building equipment hatch to the main floor over the torus	- 78.4 ft/sec	- Assume the dropped load has a diameter of 18" - RC slab, 24 inches thick - Assumes concrete will fail at approximately 1300 Kips, then calculates the penetration depth into the concrete from an initial height of 95.5 ft	- To prevent scabbing of a 24" thick floor, the missile penetration depth cannot be > 3.27" - Maximum load drop with no perforation of a 24" thick slab = 5.75 tons
Dresden 2,3	7/2/81 S&L (17)	NA	NA	95.5 ft	- Drop down the reactor building equipment hatch to the main floor over the torus	- 78.4 ft/sec	- Assume the dropped load has a diameter of 18" - RC slab, 32 inches thick - Assumes concrete will fail at approximately 1300 Kips, then calculates the penetration depth into the concrete from an initial height of 95.5 ft	- To prevent scabbing of a 32" thick floor, the missile penetration depth cannot be > 3.27" - Maximum load drop with no scabbing of a 32" thick slab = 2 tons
Dresden 2,3	7/2/81 S&L (18)	NA	NA		- Drop down the reactor building equipment hatch to the main floor, over the torus	- 78.4 ft/sec	- Assume the dropped load has a diameter of 24" - RC slab, 32 inches thick - Assumes concrete will fail at approximately 1300 Kips, then calculates the penetration depth into the concrete from an initial height of 95.5 ft	- To prevent scabbing of a 32" thick floor, the missile penetration depth cannot be > 4.62" - The amount of energy deposit to produce a penetration depth of 4.62" was calculated to be 5.36 Kips

5 SPORADIC AND INCONSISTENT 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 of Phase II by 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. As is demonstrated by Oyster Creek's proposed activities, this conclusion may not always be valid. NRC Bulletin 96-02 requested licensees to provide the staff with specific information relating to their heavy loads program and plans within 30 days. Not all licensees responded. For those licensees that did respond to the bulletin, Table 9, *Licensee response to NRC Bulletin 96-02*, provides a compilation of their responses. As shown by the table, load drop analysis, consequence analysis, plant status during load movement, and crane type to be used for the movement is incomplete.

Table 9: 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

Plant	Crane Type	Plant Status at Load Movement	Load Drop Analysis	Consequence Analysis
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
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

Plant	Crane Type	Plant Status at Load Movement	Load Drop Analysis	Consequence Analysis
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
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 SINGLE-FAILURE-PROOF CRANE CLASSIFICATION

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.

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.

Crane Classification Issues

During the information gathering phase of this candidate GI, it became clear that definite criteria did not exist for declaring a crane as single-failure-proof (e.g., for new cranes or upgraded cranes). 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 design information than NUREG-0554 in explaining design criteria for single-failure-proof cranes.

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

For example:

(1) Oyster Creek

Oyster Creek made many changes to their reactor building crane to increase its reliability, to reduce the likelihood of a load drop, and to minimize the consequence of a load drop to get approvals from the NRC to move dry storage casks at power, subsequently abandoned that approach, and installed a single-failure-proof crane in 2000.

(2) Dresden

The Dresden Unit 2/3 reactor building crane was accepted by the NRC as single-failure-proof (circa 1976) which was before NUREG-0554 was issued. Shortly after the licensing action of 1976, some of the features relied upon in classifying the crane as single-failure-proof were disabled, and they have generally remained so ever since. Current work is ongoing to restore the crane to its 1976 status, by installing new controls and limiting devices. Even when restored to the 1976 criteria, the crane will not comply with current standards as a single-failure-proof crane.

Attachment E, *Seismic Design of the Support Structures for the Reactor Building Crane*, to Sargent and Lundy calculation DRE98-0020, *Evaluation of Reactor Building Superstructure*, dated March 16, 1998, provides a summary of reactor building design criteria. It states that calculations performed in 1966-1967 showed that the stresses in the girder, the support columns and several members of the roof truss were above the materials yield stress for the (dead weight plus safe shutdown earthquake) loading. In some of the roof truss connections, the (dead weight plus safe shutdown earthquake) loading exceeded the ultimate capacity of the connections. Calculations also do not include the stresses in the support column due to the seismic (operating basis earthquake or safe shutdown earthquake) loads imposed by the siding. In 1973, the 1966-67 calculations for the crane girder and the crane columns were updated (red marked) for the effects of the new heavier single failure proof trolley. The 1973 update shows that the columns are overstressed by 6% and 35% for the operating basis earthquake and safe shutdown earthquake loading respectively, with the heavier crane trolley. In 1975 new calculations were prepared for the columns and the vertical bracing to compute the effects of the new trolley. Modifications for the columns and the vertical bracing were designed. This calculation used seismic inputs and analysis methodology more conservative than that stated in the UFSAR. The modifications designed in this calculation were not implemented. The Dresden calculation book index carries the notation “Project canceled, calculation not approved.”

Calculation DRE98-0020 (1) did not appear to include conservatisms for aging, (2) assumed that the as-built condition was the same as the design requirement, and (3) assumed that the compressive strength of affected concrete (f'_c) was 4700 psi (this value is normally assumed to be 4000 or less).

Table 10, *Dresden reactor building steel superstructure interaction summary*, shows the Interaction Coefficient (IC) for selected critical members of the reactor building superstructure where the value for the IC was 0.90 or greater. This information is shown in more detail in Section 6, *Summary and Conclusions*, of Since the IC = (actual stress)/(allowable stress), a value approaching or exceeding 1.0 may indicate an overstress situation.

Calculation DRE98-0020 provided bases for not including certain loads in the stress calculations. For example:

<u>Snow Load</u>

(The snow load is not specifically mentioned in the UFSAR.) Section 3.8.4.1.2 of the UFSAR states that the design code which was used to govern the construction documents was the Uniform Building Code (UBC), 1964 edition. The 1964 UBC did not include “live load” in the formula for total lateral load. Based on this provision, snow load is not included in the seismic loading combinations (of this calculation).

<u>Wind Load</u>

The UFSAR indicates that the wind velocity for all structures is 110 mph. Other wind loads (tornado, etc.) are not in the scope of the calculation. (Section 3.3.1.1.1 of the UFSAR indicates that the reactor building was designed to withstand winds of 170 mph, and Section 3.3.2.2.1 indicates that the reactor building is designed to withstand tornado winds to 300 mph and still safely shutdown.)

<u>Operating Basis Earthquake and Safe Shutdown Earthquake Loads</u>

The original design basis included pertinent dead and live loads as well as the OBE (or SSE) seismic loads with the crane in any location, and without lifted load. (The calculation did consider a scenario of an SSE concurrent with a maximum lifted load, but considered this to be “beyond design basis.” The calculation also does not include wind loads with either an OBE or an SSE.)

Table 10: Dresden reactor building steel superstructure interaction summary

Element description	Dead loads	Snow load	Max lifted load	Wind	OBE	SSE	IC
(1) Interior crane column member (W14x119/W24x145) (H/N/39-49)	Yes	Yes	Yes	No	No	No	0.992
(1) Interior crane column member (W14x119/W24x145) (H/N/39-49)	Yes	Yes	Yes	Yes	No	No	0.90
(2) Interior building column members (W24x145) (H/N/39-49)	Yes	Yes	Yes	No	No	No	0.996
(2) Interior building column members (W24x145) (H/N/39-49)	Yes	Yes	Yes	Yes	No	No	1.00
(3) Interior crane/building column base connections (H/N/39-49)	Yes	No	Yes	No	No	Yes	0.95
(5) Exterior column base connections (rows 38 & 50, except rows H & N)	Yes	No	No	No	No	Yes	0.97
(5) Exterior column base connections (rows 38 & 50, except rows H & N)	Yes	No	Yes	No	No	Yes	0.96
(13) Roof truss members (double angles)	Yes	No	No	No	No	Yes	0.90
(15) Roof truss members (plate girders)	Yes	Yes	Yes	No	No	No	1.05
(15) Roof truss members (plate girders)	Yes	No	No	No	Yes	No	0.95
(18) Roof truss connections (double angles)	Yes	No	No	No	No	Yes	0.90
(23) Crane girder member	Yes	Yes	Yes	No	No	No	0.93
(23) Crane girder member	Yes	Yes	Yes	Yes	No	No	0.93
(24) Crane girder connections	Yes	Yes	Yes	Yes	No	No	0.98

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 10^{-5} and 1.5×10^{-4} 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)*

EEG-74 was published by the Environmental Evaluation Group (EEG) of the New Mexico Institute of Mining and Technology in May 2000. The WIPP is located in southeast New Mexico in bedded salt at a depth of 650 meters. The repository is designed to contain 850,000 drum equivalents of contact-handled transuranic waste and 8000 canisters of remote handled transuranic waste. The contact handled waste will be shipped from various defense generator and storage sites in an NRC certified container called a TRUPACT II. The TRUDOCK system consists of two six ton cranes. Crane cable/hook breaks were initially based on relatively old (1970s) U.S. Navy data in NUREG-0612 which produced a failure rate of approximately 2.0×10^{-5} per demand. Further analysis resulted in a evaluation which produced a more realistic value of 2.5×10^{-6} per demand. The report also indicated that there was a 95% likelihood that not more than one dropped load will occur in approximately 34 years. EEG-74 is included in Appendix B.

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

This study was performed by the Office of Oversight, Office of Environment, Safety and Health, U.S. Department of Energy (DOE) in October 1996. Equipment studied included cranes, forklifts, and "other" during the period from October 1, 1993 through March 31, 1996. The "other" category included manual and power-operated hoists, chainfalls, and block and tackle. 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 10, *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. The DOE study is included in Appendix C.

Table 11: 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*

The report was prepared by the Division of Occupational Safety and Health, California Department of Industrial Relations in May 2000. Data for the report was gathered from Federal OSHA's Office of Management Data Services (OMDS) Website. Data was also gathered from Micro-to-Host reports from the Integrated Management Information System (IMIS). The report states that from January 1, 1997 through December 31, 1999, the Division of Occupational Safety and Health learned of, or had reported to it, a total of 158 accidents involving a crane. The report sorts the crane accidents by crane type, crane operator injuries, private sector vs. public sector, construction vs. non-construction, and accident causation. No mention is made concerning crane failure rates. This report is included in Appendix D.

7.5 NUREG-1738, *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*

NUREG-1738 was prepared by the Office of Nuclear Reactor Regulation and published in February 2001. This report states that for decommissioning plants:

> For a non-single-failure-proof load handling system, the drop frequency of a heavy load drop is estimated, based on NUREG-0612 information, to have a mean value of 3.4×10^{-4} per year. The number of heavy load lifts was based on the NEI (*Nuclear Energy Institute*) estimate of 100 spent fuel shipping cask lifts per year, which probably is an overestimate. For plants with a single-failure-proof load handling system or plants conforming to the NUREG-0612 guidelines, the drop frequency is estimated to have a mean value of 9.6×10^{-6} per year, again for 100 heavy load lifts per year but using data from U.S. Navy crane experience.

> Once the load is dropped, the analysis must then consider whether the drop significantly damages the SFP (*spent fuel pool*).

NUREG-1738, Appendix 2C, *Structural Integrity of Spent Fuel Pool Structures Subject to Heavy Loads Drops*, states that:

> A loss-of-inventory from the SFP could occur as a result of a heavy load drop. For single-failure-proof systems where load drop analyses have not been performed at decommissioning plants, the mean frequency of a loss-of-inventory caused by a cask drop was estimated to be 2.0×10^{-7} per year (assuming 100 lifts per year). For a non-single-failure-proof handling system where a load drop analysis has not been performed, the mean frequency of a loss-of-inventory event caused by a cask drop was estimated to be 2.1×10^{-5} per year. The staff believes that performance and implementation of a load drop analysis that has been reviewed and approved by the staff will substantially reduce the expected frequency of a loss-of-inventory event from a heavy load drop for either a single failure-proof or non-single-failure-proof system.

NUREG-1738, Appendix 2C is included in Appendix E to this report.

7.6 Navy Crane Events

NUREG-1738 used Navy crane event data from December 1995 to May 1999 to modify NUREG-0612 equations to quantify the failure rate of lifting equipment. During the time period, there were 11 incidents which involved loads in excess of 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 factors (i.e., not following procedures or lack of skills). The Navy crane event data is included in Appendix F, *Navy Crane Operating Experience*.

8 OBSERVATIONS

8.1 No Risk Significant Events Involving Cranes

There were no risk significant events involving loads of approximately 30 tons or greater at any U.S. nuclear plant having an operating license. There have been injuries and deaths caused by crane operation, but no radiation releases or risk to the health and safety of the public. There were six potentially risk-significant crane events involving a loss or partial loss of offsite power caused by mobile cranes. Two of the six events (Palo Verde and Diablo Canyon) resulted in Augmented Inspection Team (AIT) inspections, however, none of the six crane events met the minimum risk threshold requirements to be classified as an Accident Sequence Precursor (ASP) event. A review of all ASP data for the period 1985 through 1999 indicated that there were no crane events that were classified as ASP events (e.g. a minimum conditional core damage probability of 1×10^{-6} or greater).

8.2 Improving Crane Operating Performance

There were several indicators that crane operating performance has greatly improved since in issuance of NUREG-0612 in 1980. Figure 8 shows that while the number of operating plants has almost doubled, the number of load drops and load slips has remained somewhat constant. Figure 10 shows that the number of deaths and injuries dramatically decreased during post 1980 when compared to pre-1980 rates, especially when considering the rapid increase in the number of licensed operating plants. Generic Letter 85-11 also indicated 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... Therefore, a detailed Phase II review (o)f heavy loads is not necessary and Phase II is considered completed."

8.3 Inconsistent Licensee Approaches to Load Drop Calculation Methodologies, Assumptions, and Load Lift Height Restrictions

Reviews of load drop calculations obtained from each facility that was visited indicated that calculational methodologies and assumptions varied greatly from licensee to licensee, producing radically different end results. Heights of load drops, plant locations for postulated load drops, contact area at impact, materials property values, and weights of loads varied greatly. The Oyster Creek calculation for a drop of a 45 ton fuel cask over a reinforced concrete slab, 16 inches thick, was the most restrictive, with an allowable drop height of 2.77 inches. Some facilities performed load drop calculations using equations that were intended for ballistic type situations meant for high velocity and low mass. In addition, of the 74 facilities listed on Table 9, *Licensee response to NRC Bulletin 96-02*, only 8 licensee responses to Bulletin 96-02 indicated that a consequence analysis had been done at their facility for heavy load drops.

8.4 Single-Failure-Proof Crane Classification Confusion

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.

8.5 Generic Communication Documents Concerning Heavy Load Drop Issues Have Not Been Fully Effective

Despite existing NRC regulatory requirements and reminders through the generic communication process, fundamental questions still remain: (1) What is the acceptable load lift height for various loads, (2) What are the necessary crane program requirements, and (3) What are the requirements for load movements at power vs. shutdown.

Several regulatory documents have been issued that relate to very heavy loads: Unresolved Safety Issue (USI) A-36, *Control of Heavy Loads near Spent Fuel* including followup documents NUREG-0554, *Single-Failure-Proof Cranes for Nuclear Power Plants,* and NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants;* Generic Letter 80-113 (originally unnumbered), *Control of Heavy Loads;* Generic Letter 81-07, Control of Heavy Loads; Generic Letter 85-11, *Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants,* *NUREG 0612;* and Bulletin 96-02, *Movement of Heavy Loads Over Spent Fuel in the Reactor Core, or Over Safety-related Equipment.* With the exception of GL-85-11, the primary message in each of these documents was to request licensees to assess their heavy loads programs and make whatever changes that were found to be necessary.

In addition to the major heavy load documents listed above, other generic communication was issued by the NRC in addressing heavy load issues such as: IN 80-01, *Fuel Handling Events*; IN 81-23, *Fuel Assembly Damaged Due to Improper Positioning of Handling Equipment*; IN 85-12, Recent Fuel Handling Events; IN 86-06, Failure of Lifting Rig Attachment While Lifting the Upper Guide Structure At St. Lucie Unit 1; IN 86-58, *Dropped Fuel Assembly*; IN 92-13, *Inadequate Control Over Vehicular Traffic at Nuclear Power Plant Sites*; IN 96-26, *Recent Problems with Overhead Cranes*; and IN 97-51, *Problems Experienced with Loading and Unloading Spent Nuclear Fuel Storage and Transportation Casks*.

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.