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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON D.C. 20555-0001

April 11, 1996

NRC BULLETIN 96-02: MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT

Addressees

All holders of boiling-water reactor (BWR) and pressurized-water reactor (PWR) operating licenses for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to accomplish the following:

(1) Alert addressees to the importance of complying with existing regulatory guidelines associated with the control and handling of heavy loads at nuclear power plants while the plant is operating (in all modes other than cold shutdown, refueling, and defueled) and remind addressees of their responsibilities for ensuring that heavy load activities carried out under their license are performed safely and within the requirements specified under Title 10 of the Code of Federal Regulations.

(2) Request that addressees review their plans and capabilities for handling heavy loads (e.g., spent fuel dry storage casks, reactor cavity biological shield blocks) in accordance with existing regulatory guidelines [specifically NUREG-0612 (Phase I) and Generic Letter (GL) 85-11] and within their licensing basis as previously analyzed in the final safety analysis report (FSAR).

(3) Require addressees to report to the NRC whether and to what extent they have complied with the requested actions contained in this bulletin.

Although this bulletin is particularly concerned with heavy load movements while the plant is operating (i.e., in all modes other than cold shutdown, refueling, and defueled), the staff is considering further generic actions on the issue of handling all heavy loads both while the plant is operating and during shutdown.

Background

There are a number of heavy loads being handled in various areas of nuclear power plants, especially over safety-related equipment, when the plant is

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operating. Some licensees have moved or are planning to move heavy loads such as spent fuel shipping casks, transfer casks, and reactor cavity biological shield blocks during plant operations. If these loads experience uncontrolled movement or are dropped on safety-related equipment, the equipment may be unable to perform its function.

Guidelines regarding the movement of these and other heavy loads are provided in a number of documents that in combination make up the framework for the existing regulatory position on heavy load handling and control. The most important guidelines are contained in the following three documents:

(1) NUREG-0612, "Control of Heavy Loads at Power Plants," Resolution of Generic Technical Activity A-36, issued July 1980

(2) Unnumbered generic letter dated December 22, 1980, "Control of Heavy Loads"

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

NUREG-0612 provides guidelines to (1) ensure the safe handling of heavy loads, (2) reduce the potential for uncontrolled movement of heavy loads or load drops, and (3) limit the consequences of dropping a heavy load. The guidelines were supported by historical data and fault tree analyses. Some portions of the guidelines were generic to all plants, while others were specific to plant type and location (e.g., the PWR containment building). The guidelines consider the handling of heavy loads while the reactor is at power and provide a methodology to do so safely.

The unnumbered generic letter of December 22, 1980, requested that licensees implement the heavy load control guidelines in NUREG-0612 and identify any problems that they encountered. The generic letter also requested immediate implementation of some interim actions (safe load paths, crane design and inspection, operator training, and procedures), a 6-month followup response on the status of the implementation of Section 5.1.1 of NUREG-0612 (Phase I), and a 9-month followup response on the status of the implementation of the remaining applicable portions of Section 5.1 of NUREG-0612 (Phase II: single-failure-proof cranes, stops/interlocks, or load-drop analyses).

All affected licensees implemented the interim actions and Phase I of the generic letter and submitted a response for Phase II. The staff reviewed the implemented actions and a sample of the Phase II submittals and determined that the actions taken by the licensees had significantly decreased the potential for a heavy load drop. The staff performed a limited review of the remaining Phase II submittals and did not identify any plant-specific safety concerns associated with the control of heavy loads.

Subsequently, the staff issued GL 85-11, which informed licensees that implementation of Phase II was not necessary but encouraged licensees to implement any safety-significant portions they believed were appropriate. GL 85-11 relieved licensees from performing the actions requested under .



Phase II of the previous generic letter. However, GL 85-11 did not grant blanket NRC approval for all load paths identified in the Phase II submittals, nor did it authorize licensees to exceed their design basis for heavy load transfer.

Although the generic letter stated that the NRC staff review of the Phase II submittals did not indicate the need to require further generic action at that time, it did not preclude the possible future need for the staff to review additional heavy load handling concerns and to require, as appropriate, further actions by licensees.

Description of Circumstances

In 1996, GPU Nuclear (GPUN) Corporation, the licensee for the Oyster Creek Nuclear Power Plant, is scheduled to begin moving heavy loads involving dry storage casks within the Oyster Creek facility. GPUN is planning to load spent fuel from the Oyster Creek plant into dry storage casks that will be placed in an independent spent fuel storage installation. The loaded casks, each weighing 100 tons, must be moved over safety-related equipment during this process. The licensee's plans involve loading and moving the casks during power operation because performing these activities during a refueling outage would significantly increase the outage time.

The licensee prepared an initial evaluation pursuant to 10 CFR 50.59 regarding the planned activities for handling the dry storage casks, including the use of the non-single-failure-proof reactor building crane to transfer spent fuel to the dry cask storage facility during plant operation. To reduce the probability of a load drop, GPUN modified its crane; proposed to use a crush pad along part of the load path; and proposed to institute an "Error Free Plan," which includes upgrading its training, management and oversight, and cask-handling procedures specific to this evolution and development. However, during two portions of the proposed cask movement inside the reactor building, a cask drop could damage both isolation condensers and the torus, possibly creating an unisolable loss-of-coolant accident outside containment. This drop could occur in those areas near the spent fuel pool or near the equipment hatch where the crush pad proposed by the licensee to protect against drops on the 119-foot level is not installed. A cask dropped from either of these locations on the 119-foot level could fall through all of the lower floors and into the torus, damaging all equipment in its path. The licensee stated that core cooling could be maintained by steaming to the condenser using the normal feedwater system and providing makeup from the condensate storage tank and fire water systems by way of the core spray system. While GPUN had reduced the probability of dropping the cask, the staff was concerned that because the casks are heavier than previously considered in the FSAR, a cask drop could result in higher consequences than those previously analyzed.

As a result of concerns raised by the staff and GPUN's efforts to improve the efficiency of handling the spent fuel storage casks and to minimize the probability of a cask drop, GPUN updated its 10 CFR 50.59 evaluation to include a number of improvements applicable to the criteria of NUREG-0612, Phase I. GPUN adjusted the load path, eliminated the crush pad, and upgraded



the reactor building crane (but not to the level of a single-failure-proof crane as defined in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants") by installing a fixed link support system. The fixed links provide redundant rigging for the cask while it is transported on the 119-foot level, especially in the area over the isolation condensers. It uses horizontal support beams attached to the cask-lifting yoke and vertical tie-rods connected to the crane trolley to support the cask in the event of a failure of a crane hoist component.

GPUN evaluated postulated load drops while the cask is in the reactor building equipment hatchway (from the 119-foot elevation to the 23-foot elevation) and at the laydown area on the 119-foot elevation where the fixed links are not engaged and concluded that if a cask is dropped in either of these areas, the cask could damage the torus, causing it to drain. Consequently, the pressure suppression function of the primary containment could be disabled. The reactor is expected to scram successfully, reducing power so that only post-scram decay heat would have to be removed. The primary coolant system piping would not be affected by the drop; therefore, the need for vessel inventory makeup would not be required immediately. Some safety-related equipment would be damaged, for example, one set of containment spray pumps and one containment spray heat exchanger. However, containment spray would be unavailable in any event since GPUN has assumed no water would be present in the torus. The isolation condenser system would be available to provide long-term heat removal from the reactor vessel. Makeup to the isolation condenser shell could be accomplished remotely by using condensate transfer. If needed, a reactor building entry to establish shell-side makeup could be performed after approximately 1 hour. The load-drop analysis concluded that the reactor could be safely shut down following a drop of the cask and that the offsite consequences of a load drop are bounded by high-energy line break evaluations. The licensee determined that releases resulting from damage to the 52 fuel assemblies in the cask would not exceed 25 percent of the limits set out in 10 CFR Part 100 because the fuel assemblies will be more than 10 years old.

GPUN's 10 CFR 50.59 evaluation concludes that no unreviewed safety questions are involved, that movement of the casks can be accomplished in a safe manner because of GPUN's reduction of the probability of dropping the load, and that all license requirements would be satisfied. GPUN based this conclusion on its completion of the Phase I guidelines (Section 5.1.1 of NUREG-0612) for the control of heavy loads at nuclear power plants. The staff states in GL 85-11 that "our review has indicated that satisfaction of the Phase I guidelines assures that the potential for a load drop is extremely small." This conclusion is further based on GPUN's evaluation that (1) the fixed links provide redundant load support for the transfer cask, equivalent to a single-failure-proof crane for nearly the entire travel path; (2) safe shutdown can be achieved where the fixed link support system does not provide protection; and (3) although a postulated load drop could damage safety-related equipment, the probability of a drop is extremely low. The licensee also noted that the only load drop previously evaluated in the plant safety analysis report (SAR) is the drop of a 100-ton fuel shipping cask in the vicinity of the fuel pool.



Discussion

In 10 CFR 50.59(a)(1), it is stated that "the holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question." Section 50.59(a)(2) states that "a proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced."

The NRC staff audited both the initial and updated 10 CFR 50.59 evaluations performed by the licensee and determined that the proposed cask movement activities represent an unreviewed safety question that should be submitted to the NRC for review and approval pursuant to the requirements of 10 CFR 50.59 and 50.90. The staff based its determination on the fact that, as noted by the licensee, the activity involves movement of loads heavier than those previously analyzed in the FSAR (except over the cask drop protection system in the fuel pool, where a 100-ton cask drop had been previously analyzed). This determination is also based on the fact that the load drop had not been previously evaluated along the remainder of the load path, and on the possibility that a load drop in the reactor building while the reactor is at power could result in consequences that are greater than those previously postulated in the FSAR. Therefore, although the licensee had reduced the probability of dropping the cask, the staff was concerned that a load drop could result in an increase in the potential consequences. Accordingly, as defined in 10 CFR 50.59(c), if an activity is found to involve an unreviewed safety question, an application for a license amendment must be filed with the Commission pursuant to 10 CFR 50.90.

Based on the NRC staff's audit of GPUN's 10 CFR 50.59 evaluation, the staff is concerned that other licensees may believe that their heavy load operations are in compliance with the regulations because they have completed Phase I of the generic letter of December 22, 1980, and the closeout of Phase II by GL 85-11. GL 85-11 did not relieve licensees of their responsibility under 10 CFR 50.59 to evaluate new activities with respect to the SAR and the Technical Specifications to determine whether the activity involves an unreviewed safety question or a change in the Technical Specifications. In addition, GL 85-11 concluded that the risks associated with damage to safety-related systems are relatively small because (1) nearly all load paths avoid this equipment, (2) most equipment is protected by an intervening floor, (3) there is redundancy 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.



Therefore, the staff has concluded that although some licensees have under-taken efforts to further reduce the probability of an accident involving heavy loads beyond that previously accepted for NUREG-0612, Phase I, if the loads are heavier and the load paths and potential consequences of a load drop are different than those previously considered in the FSAR, the probability of an occurrence or the consequences of an accident may be increased.

Requested Actions

To ensure that the handling of heavy loads is performed safely and within the conditions and requirements specified under Title 10 of the Code of Federal Regulations, all addressees are requested to take the following actions:

- Review plans and capabilities for handling heavy loads while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) in accordance with existing regulatory guidelines. Determine whether the activities are within the licensing basis and, if necessary, submit a license amendment request. Determine whether changes to Technical Specifications will be required in order to allow the handling of heavy loads (e.g., the dry storage canister shield plug and associated lifting devices) over fuel assemblies in the spent fuel pool.

Required Response

Pursuant to Section 182a, the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), all addressees must submit the following written information:

(1) For licensees planning to implement activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment within the next 2 years from the date of this bulletin, provide the following:

   - A report, within 30 days of the date of this bulletin, that addresses the licensee's review of its 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 regulatory guidelines. The report should also indicate whether the activities are within the licensing basis and should include, if necessary, a schedule for submission of a license amendment request. Additionally, the report should indicate whether changes to Technical Specifications will be required.

(2) For licensees planning to perform activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) and that involve a potential load drop accident that has not previously been evaluated in the FSAR, submit a license amendment request in advance (6-9 months) of the planned movement of the loads so as to afford the staff sufficient time to perform an appropriate review.



(3) For licensees planning to move dry storage casks over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) include in item 2 above, 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.

(4) For licensees planning to perform activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled), determine whether changes to Technical Specifications will be required in order to allow the handling of heavy loads (e.g., the dry storage canister shield plug) over fuel assemblies in the spent fuel pool and submit the appropriate information in advance (6-9 months) of the planned movement of the loads for NRC review and approval.

Address the required written report(s) to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy of the report, to the appropriate regional administrator.

Related Generic Communications

- NUREG-0612, "Control of Heavy Loads at Power Plants," Resolution of Generic Technical Activity A-36, issued in July 1980
- Unnumbered generic letter dated December 22, 1980, "Control of Heavy Loads"
- GL 85-11: "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," June 28, 1985

Backfit Discussion

This bulletin is an information request made pursuant to 10 CFR 50.54(f). The objective of the actions requested in this bulletin is to verify that licensees are complying with the current licensing basis for their facility with respect to the proper handling and control of heavy loads at nuclear power plants when the plant is operating (in all modes other than cold shut-down, refueling, and defueled). The issuance of the bulletin is justified on the basis of the need to ensure compliance with the current licensing basis with respect to the weight of the heavy loads being moved over spent fuel, over fuel in the reactor core, or over safety-related equipment, and the potentially severe consequences that can result if a load is dropped.



Paperwork Reduction Act Statement

This bulletin contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), approval number 3150-0012, which expires June 30, 1997.

The public reporting burden for this collection of information is estimated to average 600 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. The NRC is seeking public comment on the potential impact of the collection of information contained in the generic bulletin and on the following issues:

(1) Is the proposed collection of information necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

(2) Is the estimate of burden accurate?

(3) Is there a way to enhance the quality, utility, and clarity of the information to be collected?

(4) How can the burden of the collection of information be minimized, including the use of automated collection techniques?

Send comments on any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at bjs1@nrc.gov; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0012), Office of Management and Budget, Washington, DC 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.



If you have any questions about this matter, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation

(NRR) project manager.

signed by

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Attachment: List of Recently Issued NRC Bulletins