

Affidavit of Deborah Katz

I, Deborah Katz, being duly sworn, appear and state the following:

1. I live at 80 Davenport Road in Rowe, Massachusetts. My home lies within 17 miles of the Vermont Yankee Nuclear Power Station (VYNPS)
2. I am a member of Nuclear Information and Resource Service and the President of the Citizens Awareness Network (CAN)
3. I am concerned about the effects of the experimental transfer of fuel at the Oyster Creek Nuclear Power Station while the reactor is operational into dry cask storage. The fuel will be transferred over the operating reactor vessel. It will have a direct effect on my health and safety. Since Vermont Yankee is a Mark I boiling water reactor as is the Oyster Creek reactor, the precedent that will be set by the process at Oyster Creek can directly effect me. In particular, I have the following concerns:

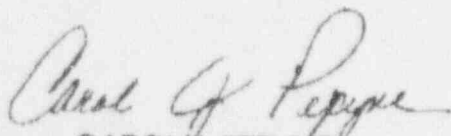
a) I travel regularly to Greenfield, MA and Brattleboro, VT which are within the 10 mile radius of the Vermont Yankee Nuclear Power Station. I am concerned that should a fuel-handling accident occur, I and my children would be exposed to unsafe radiation levels.

B) I am also concerned about the potentially devastating effects on my and my children's health and safety by a radiological accident at VYPS due to the movement of irradiated fuel in containment. We live in the effluent pathway of the reactor. Our family and our environment could be permanently contaminated by such an accident.

C) I am also concerned about the movement of fuel at VYPS since the Nuclear Engineer of Vermont, Mr William Sherman, announced at a town meeting in Buckland, MA that VYPS would begin dry cask storage of their fuel within 5 years.

I am willing to have Nuclear Information and Resource Service represent my interests at the hearing and during the intervention process.

Signed Deborah Katz Date 6/5/96



CAROL J. PEPYNE  
NOTARY PUBLIC

My Commission Expires Jan. 12, 2001

The Recorder, Greenfield, Mass., Monday, May 28, 1996

### Advocate concerned about nuclear waste

SICKLAND — Scientists have heard from Vermont state nuclear engineer William Sherman, also chairman of the Northeast High Level Radioactive Waste Task Force.

Sherman noted that his group, advocating from Delaware to Maine, has "concern about whether the federal government will ever come through with disposal" plans. It is also concerned that the Vermont Yankee Nuclear Power Plant in Vermont faces leaving spent nuclear fuel to dry cask storage — as Yankee Atomic power plant in Rowe is facing now — in about five years.

"Once it goes into dry casks, we fear it may never move again," said Sherman. "Up in Vermont we think a warm, dry place is a lot better storage than the banks of our rivers in New England."

**Attachment**

**Affidavit For Jean Burnette  
Oyster Creek Nuclear Watch  
715 Chesapeake Drive  
Forked River, NJ 08731**

**Notarized and postmarked to NRC by First Class Mail on June 5, 1996**

**Attachment**

**Affidavit for Shirley R. Schmidt**

**Oyster Creek Nuclear Watch**

**291 Wells Mill Road**

**Waretown, NJ 08758**

**609/971-6162**

**Notarized and postmarked to NRC by First Class Mail on June 5, 1996**

**Attachment**

**Affidavit of Maria Szczech  
Ocean Township Committeewoman  
Ocean Township  
50 Railroad Avenue  
Waretown, NJ 08758  
609/971-1905 609/693-3302**

**Notarized and postmarked to NRC by First Class Mail on June 6, 1996**

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

April 11, 1996

MRC 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



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 undertaken 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 shutdown, 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:

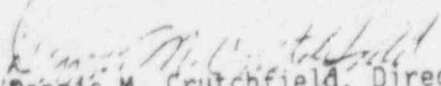
- (1) Is the proposed collection of information necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?
- (2) Is the estimate of burden accurate?
- (3) Is there a way to enhance the quality, utility, and clarity of the information 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 [bjssl@nrc.gov](mailto:bjssl@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.

  
Dennis M. Crutchfield, Director  
Division of Reactor Program Management  
Office of Nuclear Reactor Regulation

Technical contact: Brian E. Thomas, NRR  
(301) 415-1210  
Internet: bet@nrc.gov

Lead Project Manager: Kevin A. Connaughton, NRR  
(301) 415-3018  
Internet: kac@nrc.gov

Attachment: List of Recently Issued NRC Bulletins



GPU Nuclear Corporation  
 Post Office Box 388  
 Route 1 South  
 Forked River, New Jersey 08731-0388  
 609 971-4000  
 Writer's Direct Dial Number:

May 13, 1996  
 6730-96-2160

U. S. Nuclear Regulatory Commission  
 Attention: Document Control Desk  
 Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station  
 Docket No. 50-219  
 Response to NRC Bulletin 96-02.

On 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," was issued. The Bulletin contained a 30 day reporting requirement for nuclear power licensees.

After reviewing the referenced bulletin, GPU Nuclear requested and was granted by telecon a 30 day extension to submit the response. Additional time is required to perform the necessary analyses regarding lifted loads and to develop plans with regard to any license amendments which may be required. We expect to submit the response to the subject bulletin by June 10, 1996.

If any additional information is required, please contact Mr. Joseph Andrescavage of my staff at 609 971 4862.

Very truly yours,

for Michael B. Roche  
 Vice President & Director  
 Oyster Creek

200091

MBR/JFA/gl

cc: Administrator, Region I  
 NRC Project Manager  
 NRC Resident Inspector

9605200454-960513  
 PDR ADOCK 05000219  
 G PDR

JEN 1/0

c.1

UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
OFFICE OF NUCLEAR REACTOR REGULATION  
WASHINGTON, D.C. 20565

April 30, 1996

NRC INFORMATION NOTICE 96-26: RECENT PROBLEMS WITH OVERHEAD CRANES

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to recent problems with overhead cranes. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Failure of Overhead Crane Bridge Rail

At the Trojan Nuclear Plant on July 7, 1995, a section of the reactor building polar crane bridge rail failed. The rail had a crack across the top of the top flange and a piece of the flange had been displaced. The end of one section of the rail had failed through the plane of the rail joint bar bolts extending up through the top flange. Visual and metallographic examination of the failure plane indicated that much of the failure was preexisting. Rust on the failure surfaces and "peening" of some areas indicated that the initial crack could extend back to the plant's construction.

The licensee research of construction records determined that a nonconformance report, dated July 26, 1972, noted that the rails were not slotted for bolts in accordance with the drawings. The corrective action recommended was to "burn the slots in the field." The licensee determined the cause of the failure to be torsional shear and bending at the stress risers from the flame-cut holes. Flame cutting the slots left residual stresses in the material because of the lack of careful preheating and controlled cooling. Also, sharp notches, noted in the area of the flame cutting, concentrated the stresses.

The inappropriate use of a cutting torch created an untempered martensitic heat-affected zone in the high-carbon steel rail. This zone was especially sensitive to hydrogen cracking and subsequent brittle crack propagation. The crack inducing and propagating loading was primarily due to bending of the

9504260095

rail head to the outside during episodes of rail misalignment. The licensee has observed rail misalignment to be a continuing problem that had caused or contributed to 19 bridge truck wheel bearing failures over 23 years of operation.

The root cause of the failure was the inappropriate use of a cutting torch to enlarge drilled holes to slots in the web of the rail. This practice created an untempered, martensitic, heat-affected zone in the rail material that was sensitive to hydrogen cracking and subsequent brittle crack propagation.

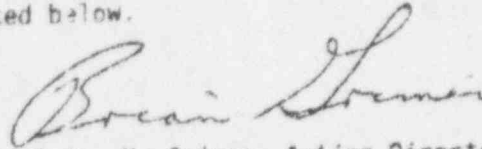
#### Actuation of Overhead Crane Safety System

At the Prairie Island Nuclear Generating Plant on May 13, 1995, while lifting a loaded spent fuel storage cask from the spent fuel pool for transfer to the transport bay, the single-failure-proof overhead crane handling system automatically stopped on overload, approximately 13 cm [5 inches] from the high hook point (peak lift point). The bottom of the cask was above the water but approximately 8 cm [3 inches] below the operating deck of the spent fuel pool. Upon investigation of the event, the licensee, Northern States Power Company (NSP) determined that the cause of the event was premature actuation of the crane overload-sensing system. The setpoint on the overload-sensing system was set too low. Upon actuation of the overload-sensing system, control power is automatically removed from the hoist motor and the conventional holding brakes are activated. Subsequent to the actuation on May 13, the cask remained in the hoisted position until a safety evaluation was made that supported bypassing the sensing system and resuming the cask lift. The lift was resumed about 16 hours after it was stopped, and the cask was placed in the decontamination area of the transport bay. NSP initiated a root-cause analysis to identify the cause of the actuation. The conclusion of this analysis was that the overload-sensing system was inaccurately calibrated.

This event raises a concern for similarly designed overload-sensing systems associated with single-failure-proof cranes. As noted in the analysis reports, this event was a "nuisance trip" that resulted from inaccurate initial calibration during load cell setting adjustment. Improved load cell accuracy can help to reduce any unbalanced loading condition in the system.

Additional details of these events can be found in Inspection Report No. 50-344/95-06 issued on September 18, 1995, and Inspection Report No. 50-282/95-06 issued on June 27, 1995.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below.



Brian K. Grimes, Acting Director  
Division of Reactor Program Management  
Office of Nuclear Reactor Regulation

Technical contacts:

Robert J. Pate, Region IV  
(510) 975-0246  
Internet: rjpl@nrc.gov

Brian E. Thomas, NRR  
(301) 415-1210  
Internet: bet@nrc.gov

David B. Pereira, Region IV  
(510) 975-0307  
Internet: dbp@nrc.gov

Eric J. Benner, NRR  
(301) 415-1171  
Internet: ejbl@nrc.gov

Russell L. Bywater, Region III  
(612) 388-8209  
Internet: rlb3@nrc.gov

Attachment: List of Recently Issued NRC Information Notices



Headquarters Daily Report  
MAY 08, 1996

Consolidated Edison Co. Of N.Y.  
Indian Point 2  
Buchanan, New York

MR Number: 1-96-0047  
Date: 05/08/96  
SRI/RI PC

Dockets: 50-247  
PWR/W-4-LP

Subject: UNIT 1 CONTAINER DROP

Discussion:

On May 7, 1996, Con Edison was in the process of moving a metal transportation container in the Unit 1, fuel handling floor. The container measured 8 ft. X 8 ft. X 20 ft. and weighed approximately 5000 lbs. Four nylon slings were used for the lift of the container. Operators attached the slings to a hook on an overhead crane. The hook had a spring loaded keeper installed to prevent the slings from sliding off. As the container was being lifted, the lighter end lifted off the floor first and caused the container to rotate. The light end of the container was lifted up approximately 18 inches when two of the slings slipped off the hook, damaging the keeper, and the container dropped to the floor.

All lifting operations on the fuel handling floor have been stopped pending review of the event.

Con Edison has determined that because the slings used were too short for this lift, the angle of the slings from the container to the hook was approximately 24 degrees. Posted guidance was to have a minimum of a 30 degrees angle to accomplish a lift. As a result, as the container rotated during the lift, the slings also rotated until they slipped off the hook. No personnel injuries or other equipment damage resulted from the drop. Con Edison is continuing to review the causes and corrective actions for this event.

Date: 12-30-94

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-II-94-055

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region II staff on this date.

FACILITY:

Georgia Power Company  
Hatch-1  
Baxley, Georgia  
Docket No. 50-321

Licensee Emergency Classification:

Notification of Unusual Event  
 Alert  
 Site Area Emergency  
 General Emergency  
 Not Applicable (EN 28194 Info Only)

SUBJECT: HATCH UNIT 1 SPENT FUEL POOL STEEL LINER PUNCTURED WHEN CORE SHROUD BOLT DROPPED

On December 28, 1994, at 8:53 p.m., Hatch Unit 1 tore a three inch diameter gash in the stainless steel liner of their spent fuel pool when a 350 pound core shroud bolt was dropped from one foot above the pool water surface. A steel cable sling failed as the bolt was being removed from the pool for shipment offsite. Seven shroud bolts had been removed from the reactor during the September 1994 refueling outage and stored in the spent fuel pool awaiting shipment offsite. Leakage through the liner gash has been contained in the annulus between the liner and the concrete outer structure of the spent fuel pool. Pool level has been restored via the normal makeup system and the falling bolt did not contact any spent fuel. The licensee is monitoring leakage of approximately 0.7 gpm which is occurring through system penetrations in the concrete structure and is being collected in the reactor building sump. There has been no release of radioactivity offsite. A contingency plug is available to insert in the gash if leakage from the concrete structure increases significantly. Contract divers are expected onsite Friday, December 30 and will assist in removal of the impacted bolt from the liner and installation of a temporary weighted gasket plug. Permanent underwater welding repairs are expected to commence Monday, January 2, 1995, at the earliest.

The Senior Resident Inspector responded onsite to monitor the licensee response at 12:15 a.m. December 29 when notified of the occurrence. The resident staff will continue to monitor licensee activities to repair the damage through the weekend.

The licensee informed the state of Georgia.

The NRC Emergency Response Center received initial notification of this event by telephone from the licensee at 2:33 a.m. (ET) on December 29, 1994.

This information is current as of 10:00 a.m. on 12/30/94.

CONTACT: S. J. Cahill - (404) 331-4198

determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to John F. Stolz: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated April 25, 1996, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the

Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Dated at Rockville, Maryland, this 3rd day of May 1996.

For the Nuclear Regulatory Commission,

**Frank Rinaldi,**

*Project Manager, Project Directorate 1-2,  
Division of Reactor Projects-I/II, Office of  
Nuclear Reactor Regulation.*

[FR Doc. 96-11431 Filed 5-7-96; 8:45 am]

BILLING CODE 7590-01-P

## **Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving no Significant Hazards Considerations**

### **I. Background**

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 13, 1996, through April 26, 1996. The last biweekly notice was published on April 24, 1996 (61 FR 18162).

### **Notice of Consideration of Issuance of Amendments To Facility Operating Licenses, Proposed no Significant Hazards Consideration Determination, and Opportunity for a Hearing**

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this

proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 7, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714



which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at a local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner

must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N102<sup>2</sup> and the following message address: to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

*Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland*

*Date of amendments request: March 28, 1996.*

*Description of amendments request: Pursuant to 10 CFR 50.90, the Baltimore Gas and Electric Company (BGE) hereby requests an amendment to Operating License Nos. DPR-53 and DPR-69 to reduce the moderator temperature coefficient (MTC) limit shown on Technical Specification Figure 3.1.1-1. This proposed change is necessary to support changes in the safety analyses made to accommodate a larger number of plugged steam generator (SG) tubes for future operating cycles. The proposed limit will be more restrictive than the existing limit to match the analytical assumptions. In addition, the licensee provided information to clarify the relationship of the MTC to an Anticipated Transient Without Scram event in its licensing basis.*

*Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:*

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The safety analyses for the current fuel cycles assume 500 tubes per steam generator (SG) are plugged and the maximum beginning-of-cycle moderator temperature coefficient (MTC) is assumed to follow the curve in Technical Specification Figure 3.1.1-1. For the fuel cycle to be installed in Unit 1 in spring 1996, Baltimore Gas and Electric Company (BGE) assumes in the analyses that more SG tubes are plugged than the current limit, and it is necessary to credit a more restrictive (less positive) limit on the maximum positive MTC to mitigate the

Testing of EDG's during power operation will not affect the availability or operation of any offsite source of power. In addition, the EDG being tested remains capable of meeting its intended design functions. Therefore the proposed change to the Technical Specification Surveillance Requirement 3.8.1.14 will not result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room*  
Location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502.  
*NRC Project Director:* William D. Beckner.

*GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey*

*Date of amendment request:* April 15, 1996 (TSCR No. 244).

*Description of amendment request:* The proposed amendment would revise Specification 5.3.1.B of the Oyster Creek Technical Specifications. The current specification prohibits handling a load greater in weight than one fuel assembly over irradiated fuel in the spent fuel storage facility. The proposed change will facilitate the off load of spent fuel to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI). Specifically, the shield plug for the dry shield canister (DSC) and the associated lifting hardware will be moved over irradiated fuel which is contained in the DSC within the transfer cask located in the Cask Drop Protection System (CDPS).

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or consequences of an accident.

The design features and capacity of the reactor building crane provide a significant safety factor. In addition, personnel training and other administrative controls further reduce risk. Thus, the dropping of the DSC shield plug onto a loaded DSC and causing damage to the spent fuel assemblies is not a

credible event. Therefore, it does not increase the probability of or consequences of an accident.

2. State the basis for the determination that the activity does or does not create the possibility of an accident or malfunction of a different type than any previously identified in the SAR [safety analysis report].

This activity will not create the possibility of a new or different type of accident than previously evaluated in the SAR because the proposed heavy load handling exception does not create a new credible accident scenario. Dropping the shield plug on a loaded DSC and damaging spent fuel assemblies is not considered a credible event.

3. State the basis for the determination that the margin of safety is not reduced.

This activity will not involve a significant reduction in the margin of safety because the proposed heavy load handling evolution does not create a credible accident scenario.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room*  
location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

*Attorney for licensee:* Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Project Director:* John F. Stolz.

*Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine*

*Date of amendment request:* April 19, 1996.

*Description of amendment request:* The proposed amendment would revise Technical Specification 5.14 to add the appropriate references identifying the detailed methodology and conditions for analyzing the Small Break Loss-of-Coolant Accident (SBLOCA) to the list of the approved Core Operating Limits Report methods.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the Proposed Amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

These Proposed Changes are administrative in nature and are consistent with the guidance set forth in the NRC Generic Letter 88-16 identifying the requirements for the inclusion of analytical methodology

references in Technical Specifications as used in determining compliance with the regulatory limits.

The references, as proposed to be included in section 5.14 of the Technical Specifications, have previously been reviewed and approved by the NRC for generic applicability to PWRs [Pressurized Water Reactors]. The reports identified in the Proposed Change have been accepted by the NRC for referencing in plant licensing applications.

Since the references listed in the Proposed Change have previously been found to meet the conditions of 10 CFR 50.46 and 10 CFR Appendix K, and that the plant specific safety analysis acceptance limits have not changed or been modified, the use of these references in the analysis of SBLOCA accident for the Maine Yankee plant is consistent with prior plant specific and industry requirements and practices.

Therefore, we have concluded that the Proposed Change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the Proposed Amendment create the possibility for a new or different kind of accident?

The Proposed Changes introduce no new mode of plant operation; do not involve the physical modification of any structure, system, or component; do not affect the function, operation or surveillance for any equipment necessary for safe operation or shutdown of the plant; and, do not involve any changes to setpoints or limits or operating parameters. The Proposed Changes are administrative in nature only.

Therefore, we have concluded that the Proposed Change cannot result in the possibility of a new or different kind of accident from that previously evaluated.

3. Does the Proposed Amendment involve a significant reduction in a margin of safety?

The Proposed Changes are administrative in nature, consistent with the guidance of Generic Letter 88-12, and have been reviewed previously by the NRC and found acceptable with regard to the requirements of 10 CFR 50.46 and 10 CFR Appendix K. Additionally, the plant specific safety analysis acceptance criteria has not changed from that used in the latest core reload analysis.

Therefore, we have concluded that the Proposed Change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room*  
location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

*Attorney for licensee:* Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011.