50-219

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 6, 1997

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO BULLETIN 96-02, "MOVEMENT OF DRY STORAGE CASKS OVER SPENT FUEL, FUEL IN THE REACTOR CORE, OR SAFETY-RELATED EQUIPMENT"

Dear Mr. Roche:

The NRC staff has evaluated the responses to Bulletin 96-02, "Movement of Dry Storage Casks over Spent Fuel, Fuel in the Reactor Core, or Safety-Related Equipment," and found that some licensees without single-failure-proof cranes have analyzed or are planning to analyze postulated spent fuel storage cask and transportation cask drop accidents to establish design basis accidents for their facilities.

Typical cask drop analyses for in-plant cask movement have addressed the effects of a drop on plant equipment and/or cask integrity. Those analyses have assumed that the cask was in its final condition with its structural lids bolted or welded in place and that the fuel remained in the cask at all times, though the integrity of the cask might be breached during the cask drop. However, since most cask lids are not secured until after the casks are removed from the pool, it is conceivable that a cask could drop in a tipped-over orientation. The cask could be also dropped back into the spent fuel pool or adjacent area, possibly dislodging the cask lid or dislodging the cask lid and ejecting some or all the spent fuel elements onto the top of the spent fuel racks, the floor of the pool, or adjacent areas.

This accident scenario involves the potential for dropping the cask during movement from the spent fuel pool to the area within the plant building where activities such as drying, inerting, and final securing of the cask lid are completed. Offsite dose effects are not expected from a cask drop and tip- over event in which there is a loss of both the cask lid and fuel confinement. However, the effect of such an event on the operation of the facility needs to be assessed. For example, evaluations may need to determine if any vital plant areas are rendered inaccessible and if operations or maintenance activities would be significantly hampered. Such evaluations would involve, but are not limited to, the cask and crane designs, the load paths, and the extent to which the licensee can demonstrate its capability of performing actions necessary for safe shutdown with resulting plant damage and in the presence of radiological source term.

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To support further NRC staff evaluation of this potential cask drop scenario while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled), please provide the following:

1. An evaluation of your crane design, load path, and cask loading and unloading processes that supports a determination that the scenario described above is not credible at your facility, or

2. If you determine that the event is credible, please provide the following:

   (a) An analysis of a possible drop of a spent fuel storage or transportation cask involving a drop that results in the tipping over of the spent fuel cask, loss of the cask lid, or loss of the cask lid and ejection of the spent fuel from the cask into the spent fuel pool or areas adjacent to the pool. This load drop/consequence analysis should include a dose analysis to personnel involved in the cask movement for the time immediately following the accident. Also, the analysis should address personnel exposure resulting from required entry into plant areas affected by the event and the impact of elevated dose fields on the ability to reach safe shutdown or continue normal plant operation.

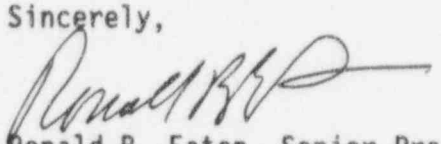
   (b) An evaluation addressing the potential for criticality resulting from the postulated cask drop accident scenario described above.

   (c) An evaluation that addresses possible means of recovering from the postulated cask drop accident scenario described above.

   (d) An evaluation that addresses whether the potential impact of the scenario described above on other parts of the facility (e.g., the spent fuel pool) is bounded by previous load drop analyses.

Please provide your response within 60 days of your receipt of this request for information. If you need clarification of the staff's request, please contact Ronald B. Eaton at (301) 415-3041.

Sincerely,



Ronald B. Eaton, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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Sincerely,
(Original Signed By)
Ronald B. Eaton, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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