GPU Nuclear

GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054
201-316-7000
TELEX 136-482
Writer's Direct Dial Number:

September 19, 1988
5000-88-1635

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Inspection 88-15
Notice of Violation Response

Attached is GPU Nuclear's response to the Notice of Violation contained in Appendix A to your letter dated August 19, 1988 which forwarded the report for the subject inspection.

If you should have any questions concerning the attachment, please contact Mr. M.W. Laggart, Manager, BWR Licensing at (201) 316-7968 or the undersigned.

Very truly yours,

R.F. Wilson
Vice President
Technical Functions

/jbw
cc: Mr. William V. Johnston, Director
Division of Reactor Safety
U.S. Nuclear Regulatory Commission - Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. William T. Russell, Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

Mr. Alex Dromerick
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

IE01

7398f

Attachment

Violation

10 CFR 50.59 requires that licensees obtain Commission approval for facility changes that involve an unreviewed safety question. An unreviewed safety question, as defined in 10 CFR 50.59, is an increase in the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in a safety analysis report.

Contrary to the above, the licensee performed an evaluation per 10 CFR 50.59 that accepted a calculated reduction in stress margins, below levels previously established by the NRC, which represented an increase in failure probability for two piping systems attached to the containment suppression pool. Although more sophisticated calculations performed subsequent to the NRC inspection demonstrated that the piping stresses are below established allowable values, acceptance of the original stress calculations without NRC approval was not in accordance with the provision of 10 CFR 50.59 as stated above.

Response

GPU Nuclear does not concur with the violation stated above.

The applicable safety analysis report is the NRC staff Safety Evaluation (SE) dated January 13, 1984, which included Franklin Research Center Technical Evaluation Report (TER) C5506-319 dated August 30, 1983. The TER documented the acceptability of calculated overstress at certain torus-attached-piping locations documented in the original plant unique analysis report (MPR-734) dated August 1982. The acceptable basis considered the following conservatisms:

1) Modal responses were combined by absolute summation.

2) Response spectrum dynamic loads were peak-broadened $\pm$ 10%.

3) The response spectrum method used provides up to 30% higher stress than time-history analysis.

When intended modification plans to torus-attached-piping were changed subsequent to the issuance of the NRC staff's SE, reanalyses of some torus-attached-piping systems were necessary as the analysis results in MPR-734 assumed those modifications would be implemented. The results of the reanalyses were then documented in a followup report (MPR-999). At two locations, calculated stresses slightly exceeded Code allowables at a similar magnitude as those documented in MPR-734. Since modal responses were combined using square-root-of-the-sum-of-the-squares versus absolute summation during the reanalysis, one element of conservatism was removed. However, the remaining elements of known conservatism were judged to be sufficient to conclude that no actual overstress condition would exist for the applicable loading combinations. With this understanding, the question "Is there an increase in the probability of a malfunction of equipment important to safety previously evaluated in a safety analysis report increased" can be answered no.

GPUN acknowledges that engineering judgement was used in the 10 CFR 50.59 determination versus reliance on explicit Code-conforming design values. However, we believe that supportable engineering judgement which is adequately documented and prudently applied with appropriate conservatism is acceptable for 10 CFR 50.59 reviews.

In response to questions raised during the inspection, GPUN submitted a letter dated June 1, 1988 which provided the detailed discussion of the conservatisms inherent in the analysis methodology. Because NRC questioned the conclusion of our engineering judgement, GPUN also directed its consultant to reanalyze the two piping systems where the calculated overstress conditions were located. For the 6 inch south containment spray test return branch connection, piping system flexibility was more realistically modeled and the stresses calculated were less than the allowables. A time-history analysis of loadings to the 6 inch nitrogen purge branch connection was performed with results indicating about a 50% stress reduction from the results documented in MPR-999, Revision 1. The time-history analysis confirms GPUN's judgement regarding the calculated overstresses.

As identified at our July 26, 1988 meeting at Region I offices, GPUN, in retrospect, should have formally submitted MPR-999, Revision 1 to fully inform NRC of all analyses and modifications performed as a result of the Mark I Containment Long-Term Program. A review of Mark I Containment Long-Term Program analyses indicated no other cases of calculated overstress conditions which are unapproved by the NRC. MPR-999, Revision 1 will be submitted shortly under separate cover. Revision 2 to MPR-999 will be submitted, when available, to document the analyses performed as a result of the questions raised by the NRC during inspection 88-15.