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Vice President
Nuclear Operations

May 10, 2001

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I
Washington, D.C. 20555

Subject: Containment Pressure
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

In 1999, RG&E received a notice from Westinghouse regarding a non-conservatism in their steam line break methodology. As a result, RG&E's evaluation of a steam line break with a limiting single active failure of the feedwater regulation valve to close indicated that containment pressure would increase above the Ginna 60 psig design pressure. Corrective action included cycle-specific use of reactivity feedback coefficients, as well as a limitation on our use of the full range of currently analyzed average Reactor Coolant System average temperature.

As a result of the above changes, the analyzed post-steam line break pressure is below 60 psig, but RG&E has determined that it would be a significant benefit for Ginna Station to regain the operational flexibility relative to these two parameters (reactivity feedback coefficients and RCS average temperature). This could be accomplished by increasing the allowable pressure in containment following a steam line break from 60 psig to 69 psig. The 10CFR50, Appendix J, Option B and the Ginna Station Technical Specification definition of design pressure would however not change and would remain at 60 psig.

The containment is not considered a fission product barrier for a steam line break. The radiological consequences of a steam line break are minimal, since fuel damage does not occur and the Reactor Coolant System remains intact. When the dose consequences of a steam line break were evaluated (SEP Topic XV-2 SER dated 9/24/81), the analysis was done for a steam line break <u>outside</u> containment, since this limiting case would result in a greater calculated release. Even so, the offsite consequences were calculated to be 0.1 rem whole body and 62 rem thyroid, well within the 10CFR Part 100 guidelines.

For steam line breaks inside containment, the containment must remain intact, so as not to cause any consequential damage to safety-related equipment used to mitigate a steam line break or to exacerbate its effects. In order to ensure this structural integrity, RG&E plans to perform an analysis using the ANSYS code to demonstrate that containment remains within its code-

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allowable limits at this higher pressure. RG&E also performed an elevated containment pressure test up to 72 psig in 1996 following Steam Generator Replacement. Strain test results taken during that test were very positive.

RG&E has initially determined that this change meets the definition of an activity that results in a design basis limit for fission product barrier as described in the UFSAR being exceeded or altered under 10CFR50.59 (c)(2)(vii) requiring NRC review and approval. RG&E would like to meet with the affected review branches to discuss our approach, the timing of the analysis, and any required RG&E submittals. A meeting by the end of May would be appreciated.

Very truly yours,

Robert C. Mecredy

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xc: Mr. Guy S. Vissing (Mail Stop 8C2)
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